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NUREG-0857
Supplement No. 9

# Safety Evaluation Report

## related to the operation of Palo Verde Nuclear Generating Station, Units 1, 2, and 3

Docket Nos. STN 50-528, STN 50-529, and STN 50-530

Arizona Public Service Company, et al.

**U.S. Nuclear Regulatory Commission**

**Office of Nuclear Reactor Regulation**

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## ABSTRACT

Supplement No. 9 to the Safety Evaluation Report for the application filed by Arizona Public Service Company et al. for licenses to operate the Palo Verde Nuclear Generating Station, Units 1, 2 and 3 (Docket Nos. STN 50-528/529/530), located in Maricopa County, Arizona, has been prepared by the Office of Nuclear Reactor Regulation of the Nuclear Regulatory Commission. The purpose of this supplement is to update the Safety Evaluation Report by providing an evaluation of (1) additional information submitted by the applicant since Supplement No. 8 was issued and (2) matters that the staff had under review when Supplement No. 8 was issued, specifically those issues which required resolution prior to Unit 2 fuel loading and testing up to 5% of full power.




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(2) Environmental qualification - The condition has been changed to reflect the Commission's granting of an extension to March 30, 1986, for qualification of the hydrogen recombiners.

(3) Supplement 1 to NUREG 0737 requirements -

a. Procedures generation package: By letter dated July 10, 1985 the applicant committed to upgrade the Emergency Operating Procedures of Unit 2 to Revision 2 of CEN-152. This commitment has been fulfilled and was documented by the applicant in a letter dated November 8, 1985. Hence, this portion of the condition has been met.

b. Detailed Control Room Design Review: This portion of the condition has been revised to require APS, prior to exceeding 5% of full power, to identify any differences between the Unit 1 and Unit 2 control rooms, to evaluate the differences to identify HEDs, if any, and to propose actions and schedules to correct the HEDs.

c. Human Engineering Discrepancy Corrections: This portion of the condition has been revised to reflect the applicant's November 8, 1985 submittal on HED correction status.

d. Post accident dose assessment: This portion of the license condition has been satisfied for all three units by the applicant's letter dated June 27, 1985.

e. Safety Parameter Design System: The applicant submitted a schedular commitment in letter dated November 8, 1985. The applicant stated that the SPDS for both Units 1 and 2 will be ready for an NRC audit by April 30, 1986. This portion of the license condition has been revised to reflect this submittal.

f. Regulatory Guide 1.97, Revision 2: By letter dated November 8, 1985, the applicant submitted a schedule for testing R.G. 1.97 instruments. An appropriate condition has been included in the license.

The following license condition has been added to the Unit 2 low power license.

(1) Inservice inspection program - This condition requires that the applicant submit a Unit 2 inservice inspection program for staff approval, nine months after the issuance date of the Unit 2 low power license.



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# 3 DESIGN CRITERIA - STRUCTURE, COMPONENTS, EQUIPMENT AND SYSTEMS

## 3.10 Seismic and Dynamic Qualification of Safety-Related Mechanical and Electrical Equipment

Evaluation of the applicant's program for seismic and dynamic qualification of safety-related electrical and mechanical equipment consists of: (1) a determination of the acceptability of the procedures used, standards followed, and the completeness of the program in general, and (2) an audit of selected equipment items to develop a basis for the judgement of the completeness and adequacy of the implementation of the entire seismic and dynamic qualification program.

Guidance for the evaluation is provided by the Standard Review Plan (SRP) Section 3.10, and its ancillary documents, Regulatory Guides (R.G.) 1.61, 1.89 and 1.92, 1.100, NUREG-0484, and Institute of Electrical and Electronics Engineers (IEEE) Standards 344-1975 and 323-1974. These documents define acceptable methodologies for the seismic qualification of equipment. Conformance with these criteria is required to satisfy the applicable portions of the General Design Criteria (GDC) 1, 2, 4, 14, and 30 of Appendix A to 10 CFR Part 50, as well as Appendix B to 10 CFR Part 50 and Appendix A to 10 CFR Part 100.

Evaluation of the program at Palo Verde Unit 1 was performed by a Seismic Qualification Review Team (SQRT) and a Pump and Valve Operability Review Team (PVORT) which consisted of engineers from the Equipment Qualification Branch (NRC/EQB) and the Idaho National Engineering Laboratory (INEL, EG&G Idaho). The SQRT and PVORT reviewed the equipment qualification information contained in the Final Safety Analyses Report (FSAR) Sections 3.9.3.2 and 3.10 and made a plant site visit from August 3 through August 6, 1982 and from February 15 through February 17, 1983. The purpose of the site visit was to determine the extent to which the qualification of equipment, as installed, meets the criteria described above. A representative sample of safety-related electrical and mechanical equipment, as well as instrumentation, included in both Nuclear Steam Supply System (NSSS) and Balance of Plant (BOP) scopes, was selected for the audit. The plant site visit consisted of field observations of the actual, final equipment configuration and its installation. This was followed by a review of the corresponding design specifications, test and/or analysis documents which the applicant maintains in his central files. Observing the field installation of the equipment is necessary to verify and validate equipment modeling employed in the qualification program. In addition to the document reviews and equipment inspections, the applicant presented details of the maintenance, startup testing, and in-service inspection programs.

The purpose of this SER is to evaluate the adequacy of the Palo Verde Unit 2 equipment qualification program for safety-related mechanical and electrical equipment. In order to document the degree to which the equipment qualification program complies with the qualification requirements and criteria, the applicant provided equipment qualification information by letter dated October 23, 1985.



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The scope of this report is limited to an evaluation of the safety-related mechanical equipment and electrical equipment at Palo Verde Unit 2 that is different from equipment at Unit 1 and which must function in order to mitigate the consequences of design basis accident, inside or outside containment, while subjected to the full range of normal and accident loadings (including seismic).

Safety-related mechanical equipment and electrical equipment at Palo Verde Unit 2 that is identical to equipment at Unit 1 were addressed in SER Supplements 5 and 7 to NUREG-0857.

By letter dated October 23, 1985, the applicant has confirmed that the safety-related mechanical equipment and electrical equipment types and locations for Unit 2 are identical to Unit 1 except for two components. Hence, SER supplements 5 and 7 to NUREG-0857 addressing the qualification of equipment are also applicable to Palo Verde Unit 2. The applicant has stated that the components which are different are qualified for their application. These two items are Valcor solenoid valves located in the auxiliary pressurizer spray lines and ITT Barton pressure transmitters which provide interlock actuation for the shutdown cooling system isolation valves.

The staff has reviewed the Palo Verde Unit 2 program for seismic and dynamic qualification of mechanical and electrical equipment that is safety related. The staff finds the seismic and dynamic qualification program for safety-related mechanical and electrical equipment at Palo Verde Unit 2 meets the applicable portions of GDC 1, 2, 4, 14, and 30 of Appendix A to 10 CFR Part 50, Appendix B to 10 CFR Part 50, and Appendix A to 10 CFR Part 100.

## 3.11 Environmental Qualification of Electric Equipment Important to Safety and Safety-Related Mechanical Equipment

Equipment which is used to perform a necessary safety function must be demonstrated to be capable of maintaining functional operability under all service conditions postulated to occur during its installed life for the time it is required to operate. This requirement, which is embodied in General Design Criteria 1 and 4 of Appendix A and Sections III, XI, and XVII of Appendix B to 10 CFR 50, is applicable to equipment located inside as well as outside containment. More detailed requirements and guidance relating to the methods procedures for demonstrating the capability for electrical equipment have been set forth in 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plant"; NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," which supplements IEEE Standard 323; and various NRC Regulatory Guides and industry standards.

NUREG-0588 was issued in December 1979 to promote a more orderly and systematic implementation of equipment qualification programs by industry and to provide guidance to the NRC staff for is use in ongoing licensing reviews. The positions contained in that report provide guidance on (1) how to establish environmental service conditions, (2) how to select methods which are considered appropriate for qualifying equipment in different areas of the plant, and (3) other areas such as margin, aging and documentation for each item of safety-related electrical equipment and to identify the degree to which their qualification programs complied with the staff positions discussed in NUREG-0588.



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IE Bulletin 79-01B, "Environmental Qualification of Class IE Equipment," issued January 14, 1980, and its supplements dated February 29, September 30, and October 24, 1980, established environmental qualification requirements for operating reactors. This bulletin and its supplements were provided to OL applicants for consideration in their review.

A final rule on environmental qualification of electrical equipment important to safety for nuclear power plants became effective on February 22, 1983. This rule, Section 50.49 of 10 CFR 50, specifies the requirements to be met for demonstrating the environmental qualification of electrical equipment important to safety located in a harsh environment. In accordance with 10 CFR 50.49, electrical equipment in the Palo Verde may be qualified in accordance with the acceptance criteria specified in Category I of NUREG-0588.

In order to document the degree to which the environmental qualification program complies with the NRC's environmental qualification requirements and criteria, the applicant provided equipment qualification information by letter dated October 7, 1985.

The purpose of this SER is to evaluate the adequacy of the Palo Verde environmental qualification program for safety-related mechanical equipment and electrical equipment important to safety as defined in 10 CFR 50.49. The staff position relating to open items, as well as any unresolved issues, is provided in this report.

The scope of this report is limited to an evaluation of the safety related mechanical equipment and electrical equipment important to safety at Palo Verde Unit 2 that is different from equipment at Unit 1 and which must function in order to mitigate the consequences of a design basis accident, inside or outside containment, while subjected to the hostile environment associated with these accidents.

Safety-related mechanical equipment and electrical equipment important to safety at Palo Verde Unit 2 that is identical to equipment at Unit 1 were addressed in SER Supplements 5, 7, and 8 (NUREG-0857).

By letters dated October 7 and October 23, 1985, the applicant has confirmed that the safety-related mechanical equipment and electrical equipment important to safety types and locations for Unit 2 are identical to equipment in Unit 1 except for two components. These two components are Valcor solenoid valves located in the auxiliary pressurizer spray lines and ITT Barton pressure transmitters which provide interlock actuation for the shutdown cooling system isolation valves. The applicant has stated that the components which are different than the Unit 1 components are qualified for their application. Hence, SER Supplements 5, 7, and 8 (NUREG-0857) addressing the environmental qualification of electrical equipment are also applicable to Palo Verde Unit 2. However, in their letter of September 30, 1985, the applicant has identified that the Hydrogen Recombiners are not yet qualified and requested an extension until March 30, 1986 to complete the qualification of this equipment. By letter dated November 18, 1985, the Commission approved this request. Hence a license condition to that effect will be incorporated into the Palo Verde Unit 2 license.

The staff has reviewed the Palo Verde Unit 2 program for environmental qualification of electrical equipment important to safety and safety-related mechanical



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equipment. As noted above, this review is limited to equipment in Palo Verde Unit 2 that is within the scope of 10 CFR 50.49 and safety-related mechanical equipment that is different from equipment in Palo Verde Unit 1. The purpose of the review was to assess the qualification status of such equipment and to determine the adequacy of the qualification program. The staff's review has determined that the following license condition should be included in the Palo Verde Unit 2 license. (The referenced Commission letter is included as Appendix K to this SSER.)

> Pursuant to the extension granted in the Commission letter of November 18, 1985, APS shall environmentally qualify the hydrogen recombiners according to the provisions of 10 CFR 50.49 by March 30, 1986.

Based on the results of our review and evaluation, the staff concludes that the applicant has demonstrated compliance with the requirements of 10 CFR 50.49, the relevant parts of General Design Criteria 1 and 4 of Appendix A and Sections III, XI, and XVII of Appendix B to 10 CFR 50, and the criteria specified in NUREG-0588.



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# 4 REACTOR

## 4.2 Fuel System Design

### 4.2.5 Guide Tube Wear Surveillance

In the CESSAR SER, the staff stated that the first two CESSAR applicants were required to have fuel assembly guide tube wear surveillance programs. Palo Verde Units 1 and 2 are the first two CESSAR plants. The staff has closed out this issue for Unit 1 based on the applicant's commitment to implement an inspection program 6 months before the first refueling outage.

By a letter dated August 12, 1985, from E. E. Van Brunt, Jr. (ANPP), to G. W. Knighton (NRC), the applicant stated that a similar inspection program for Unit 2 will be implemented 6 months before the first refueling outage. However, the scope of the Unit 2 program will be defined when the Unit 1 inspection and data analysis are complete. The staff considers this an acceptable alternative.

The staff therefore concludes that the applicant's commitment to implement an inspection program is acceptable, and the issue of guide tube wear surveillance can be closed out for Unit 2.


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# 5 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

## 5.2 Integrity of Reactor Coolant Pressure Boundary

### 5.2.4 Reactor Coolant Pressure Boundary Inservice Inspection and Testing

This evaluation provides additional conclusions to information in this section of the SER and Supplements 1 and 3, which address the definition of examination requirements and the evaluation of compliance with 10 CFR 50.55a(g).

#### 5.2.4.1 Evaluation of Compliance for Palo Verde Unit 1 With 10 CFR 50.55a(g)

The SER states that the initial inservice inspection (ISI) program had not been submitted for review. In a letter dated August 26, 1985, the applicant submitted the Unit 1 ISI program manual for review and approval. This document describes the ASME Code* Class 1, 2 and 3 components and their supports selected by the applicant for examination based on ASME Code Section XI during the initial 10-year inspection interval. The ISI program also includes additional examinations that are not required by ASME Code Section XI, such as the following:

(1) Surveillance requirements for the reactor coolant pump flywheels defined in the Technical Specifications

(2) Commitments to perform inspections based on IE Bulletins 79-13, 80-27, 82-02 and 82-09

(3) Augmented examination of high energy fluid system piping

(4) Augmented examination of the residual heat removal (RHR), emergency core cooling (ECC) and containment heat removal (CHR) systems

The applicant has requested written relief from requirements that the applicant determined to be not practical to perform in accordance with 10 CFR 50.55a(g)(5)(iii).

These requests for relief include the following subjects:

(1) The inservice functional testing of hydraulic and mechanical snubbers in accordance with ASME Code Section XI Article IWF-5000

(2) The volumetric examination of pressure vessel nozzle inner radius sections

(3) The removal of insulation associated with the visual examination of mechanical or welded attachments to pressure retaining components

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*The Boiler and Pressure Vessel Code of the American Society of Mechanical Engineers.



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The staff evaluation of the applicant's August 26, 1985 letter is continuing because additional information is required to complete the review. Based on the available information, the staff has reached a preliminary conclusion that the sample of welds and component supports selected by the applicant for inservice examination significantly exceeds the requirements of ASME Code Section XI, 1980 Edition, including addenda through Winter 1981.

5.2.4.2 Evaluation of Compliance for Palo Verde Units 2 and 3 With 10 CFR 50.55a(g)

SSER 3 states that the applicant was considering updating the preservice inspection (PSI) program to meet later requirements of ASME Code Section XI. In a letter dated February 5, 1985, the applicant provided a comparison of the PSI programs for all units and information specific to Units 2 and 3. The PSI programs for PVNGS 1-3 are similar to the programs for other Combustion Engineering (CE) plants that have been docketed and consist of shop and field examinations. The principal components of the reactor coolant pressure boundary, such as the reactor vessel, steam generators, pressurizer, and reactor coolant pumps, were examined in the fabrication shop. Certain welds in these principal components and welds in the primary piping system were conducted during the field preservice examination.

In the comparison the applicant states that the shop PSI programs for PVNGS 1-3 were conducted in accordance with the 1974 Edition, Summer 1975 Addendum of ASME Code Section XI. All three programs were essentially identical; with no significant differences between the units. The Field PSI programs for Units 2 and 3 were initially developed in accordance with the 1974 Edition, Summer 1975 Addendum of ASME Code Section XI; however, these programs have been revised to comply with the requirements of the 1977 Edition, Summer 1979 Addendum of Section XI, subject to the limitations of 10 CFR 50.55a(b) regarding the adoption of this Code edition. The only exception is that Class 1, 2, and 3 component support examinations will remain subject to the 1974 Edition, Summer 1975 Addendum. The only programmatic difference between the Unit 1 field program, with its associated relief requests, and the Units 2 and 3 programs is a change from volumetric to surface examination techniques for Class 1 circumferential welds on less than 4-inch nominal piping systems.

The applicant provided plant specific information as follows:

(1) Multi-unit Calibration Blocks

The PVNGS 2-3 shop PSI programs utilize the same ultrasonic testing calibration blocks for the major components of the nuclear steam supply system (NSSS) as was used in the Unit 1 PSI. The Units 2 and 3 NSSS components have been fabricated to the same specification, product form, and specified heat treatment as the Unit 1 components. Thus, each calibration block is of a material of the same specification, product form, and heat treatment as one of the materials being joined, which is in accordance with the requirement set forth in the Summer of 1979 Addendum to ASME Code Section XI. Although the calibration blocks are not from actual component dropouts or prolongations, their usage will result in acceptable examination results consistent with current industry practice and code requirements.



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(2) Uninspectable Regions of Reactor Vessel Welds

With the exception of the electronic gating effects associated with the inside diameter (ID) examination of the reactor vessel, all the pressure retaining welds in the reactor were examined during the shop PSI program for the Units 2 and 3 components. The effect of the electronic gating characteristics experienced during the reactor vessel examinations resulted in approximately 0.6 to 0.8 inch of metal on the ID surface not being examined during the 0° scans and approximately 0.2 to 0.3 inch not being examined during the 60° angle beam scans.

In SSER 1 the staff found acceptable the selection of the primary pressure boundary welds subject to examination in the Unit 1 PSI program. The regulation permits all components (including supports) to meet the requirements in the subsequent ASME Code editions and addenda referenced in 10 CFR 50.55a(b). The staff has determined that the applicant used acceptable methods to update the PSI programs for Units 2 and 3. Therefore, the staff concludes that the sample of primary pressure boundary welds subject to PSI in Units 2 and 3 is acceptable.

The staff determined that the ultrasonic testing calibration blocks used during the PSI of Units 2 and 3 meet the material specification requirements of later approved Codes and, therefore, these calibration standards are acceptable. The preservice examination of the reactor vessels at PVNGS was conducted in the fabrication shop, using approved editions of ASME Code Section XI, prior to the publication of augmented requirements described in Regulatory Guide (RG) 1.150, "Ultrasonic Testing of Reactor Vessel Welds During Preservice And Inservice Examination," Revision 1. The staff considers the preservice examination of the Units 2 and 3 reactor vessel acceptable because the examinations were consistent with the applicable Code and the commercial practice at the time of examination. The limitation to examination resulting from electronic gating during the PSI should be significantly reduced during future inservice inspections by the implementation of RG 1.150 based on Generic Letter 83-15, which was sent on March 23, 1983 to all licensees of operating power reactors and applicants for operating licenses.

The applicant requested relief from certain Code preservice requirements that the applicant determined to be impractical to perform on Unit 2 and provided a supporting justification in letters dated February 5, 1985 and July 5, 1985. The technical issues are similar to those identified on Unit 1 and addressed in SSER 3. The staff evaluated the ASME Code-required examinations that the applicant determined to be impractical and the staff determined that the applicant has demonstrated that either (1) the proposed alternatives would provide an acceptable level of quality and safety or (2) compliance with the requirements would result in hardships or unusual difficulties without a compensating increase in the level of quality and safety. Based on the review of the applicant's submittals and the granting of relief from these preservice examination requirements, the staff concludes that the preservice inspection program for PVNGS Unit 2 is acceptable and in compliance with 10 CFR 50.55a(g)(3). The detailed evaluation supporting this conclusion is provided in Appendix L to this report. The staff will evaluate requests for relief from impractical examinations on Unit 3 after detailed plant-specific information is provided



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by the applicant. The staff will report its conclusions in a future supplement to the SER.

## 5.3 Reactor Vessel

### 5.3.1 Reactor Vessel Materials

The acceptance criteria and references that are the basis for this evaluation were set forth in the SER.

Compliance With Appendix G, 10 CFR 50

Appendix G requires that, for the reactor beltline materials, the Charpy V-notch ($C_v$) impact tests shall be conducted at appropriate temperatures over a temperature range sufficient to define $C_v$ test curves (including the uppershelf levels) in terms of both fracture energy and lateral expansion of specimens. The applicant provided the impact test data in FSAR Amendments 12 and 14. The staff has reviewed these data to determine the degree of compliance with the fracture toughness requirements of Appendix G to 10 CFR 50. Based on its review, the staff concludes that PVNGS 2 and 3 have met all the requirements of Appendix G except that the applicant has not provided pressure-temperature limit curves for Unit 3. This is further discussed in Section 5.3.2 below.

Compliance With Appendix H, 10 CFR 50

The materials surveillance programs at PVNGS 2 and 3 will be used to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region, resulting from exposure to neutron irradiation and the thermal environment as required by General Design Criterion (GDC) 32, "Inspection of Reactor Coolant Pressure Boundary." The PVNGS 2 and 3 surveillance programs must be in compliance with Appendix H to 10 CFR 50 and ASTM E-185, "Standard Recommended Practices for Surveillance Tests for Nuclear Reactor Vessels," require that fracture toughness data be obtained from material specimens that are representative of the limiting base, weld and heat-affected zone materials in the beltline region. These data will permit the determination of the conditions under which the vessel can be operated with adequate margins of safety against fracture throughout its service life.

Based on a review of the applicant's submittal, the staff has determined that the PVNGS 2 and 3 reactor vessel surveillance programs meet the requirements of Appendix H of 10 CFR 50 and ASTM E-185.

### 5.3.2 Pressure Temperature Limits

The acceptance criteria and references that are the basis for this evaluation, were set forth in the SER.

Appendix G, "Fracture Toughness Requirements," and Appendix H, "Reactor Vessel Material Surveillance Program Requirements," 10 CFR 50, describe the conditions that require pressure-temperature limits and provide the general bases for these limits. These appendices specifically require that pressure-temperature limits must provide safety margins at least as great as those recommended in Appendix G to ASME Code Section III, "Protection Against Non-Ductile Failures."



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Appendix G to 10 CFR 50 requires additional safety margins for the closure flange region materials and beltline materials whenever the reactor core is critical, except for low-power physics tests.

The following pressure-temperature limits imposed on the reactor coolant pressure boundary during operation and tests are reviewed to ensure that they provide adequate safety margins against non-ductile behavior or rapidly propagating failure of ferritic components as required by GDC 31.

(1) Preservice hydrostatic tests
(2) Inservice leak and hydrostatic tests
(3) Heatup and cooldown operations
(4) Core operation

The applicant has provided pressure-temperature limits for Unit 2 but not for Unit 3. The staff has reviewed the pressure-temperature limits for Unit 2 and has determined that they comply with the requirements of Appendix G, 10 CFR 50. The applicant indicates that the pressure-temperature limits for Unit 3 will be issued as part of that unit's Technical Specifications. Hence, the staff will review the pressure-temperature limits for Unit 3 together with the Technical Specifications for Unit 3.

The pressure-temperature limits to be imposed on the reactor coolant system for all operating and testing conditions must be in conformance with established criteria, codes, and standards. The use of operating limits based on these criteria, as defined by applicable regulations, codes and standards, will provide reasonable assurance that non-ductile or rapidly propagating failure will not occur, and will constitute an acceptable basis for satisfying the applicable requirements of GDC 31.

5.3.3 Reactor Vessel Integrity

The acceptance criteria and references that are the basis for this evaluation, were set forth in the SER. Although most areas are reviewed separately in accordance with other review plans, reactor vessel integrity is of such importance that a special summary review of all factors relating to reactor vessel integrity is warranted. Hence, the staff has reviewed the fracture toughness of ferritic reactor vessel and reactor coolant pressure boundary materials, the pressure-temperature limits for operation of the reactor vessel, and the materials surveillance program for the reactor vessel beltline. The staff has reviewed the information in each area to ensure that it is complete and that no inconsistencies exist that would reduce the certainty of vessel integrity. The areas reviewed and the SER sections containing the review results are:

(1) Design (5.3.1)
(2) Materials of construction (5.3.1)
(3) Fabrication methods (5.3.1)
(4) Operating conditions (5.3.2)

The staff has reviewed these factors contributing to the structural integrity of the reactor vessel and concludes that the applicant has complied with Appendices G and H of 10 CFR 50 and there are no special considerations that make it necessary to consider potential reactor vessel failure for PVNGS 2 and 3.


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## 5.4 Component and Subsystem Design

### 5.4.3 Shutdown Cooling (Residual Heat Removal) System

In SSER 8, the staff indicated that the applicant's revised boron mixing and natural circulation cooldown test procedure was under staff review and that staff comments would be provided so that an acceptable test procedure could be finalized. The staff concerns regarding the test procedure were transmitted to the applicant during the June 27, 1985, meeting and telecon of July 10, 1985. In response the the applicant stated in its letter dated August 29, 1985 (ANPP-33282) that the following changes have been made to the procedure:

(1) The procedure has been modified to assume a single failure. Only one train of safety related equipment will be used during the test.

(2) The letdown portion of the Chemical and Volume Control System (CVCS) will be secured during the portion of the depressurization where the auxiliary pressurizer spray system (APSS) is utilized.

(3) The pressurizer heaters will not be utilized during the test. However, the pressurizer heaters will be available for use during the test.

(4) Only two of the three charging pumps will be used during the test. However, the third charging pump will be available for use if necessary.

The staff has completed it's review of the procedure, as modified, and finds that it is acceptable for the conduct of the test. Following completion of the test, the applicant should submit a post-test report which should include an evaluation of how the plant's performance is in conformance with BTP RSB 5-1.

The applicant had proposed to delay the natural circulation cooldown test until Spring of 1986 based on providing more operational flexibility and more decay heat during the test. In evaluating this proposal, the staff notes the following:

(1) The Palo Verde reactor vessel upper head (RVUH) is the largest of any U. S. reactor. This large RVUH significantly slows the rate of system depressurization.

(2) A natural circulation cooldown has never been attempted on the System 80 design.

(3) The cooldown process would require a RVUH fill and drain procedure which has never been attempted on the System 80 design.

(4) The natural circulation cooldown test will also test the APSS flow characteristics with a check valve removed from main spray line. This arrangement has never been tested.

(5) Palo Verde Unit 1 has demonstrated some unreliability in its offsite power system (due to multiplexer problems). Natural circulation cooldown will provide greater assurance of the ability of the CE System 80 design to safely reach cold shutdown following a loss of offsite power.



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(6) Palo Verde Unit 1 has demonstrated some unreliability of the safety-grade charging pumps due to hydrogen gas binding. The natural circulation cooldown test would verify the ability to avoid gas binding.

Based on these concerns, the staff believed that the proposed delay was not justified. The added decay heat as a result of delaying the test until Spring 1986, while making the test more representative of an actual natural circulation cooldown procedure throughout the facility's operating life, does not add significantly to the value of the test in terms of ensuring the plant's ability to accomplish the required cooldown. For these reasons, the staff informed the applicant that the proposed delay would not be acceptable, and the staff requested an alternative schedule.

In response to the staff request, the applicant committed, in its letter dated November 27, 1985 (ANPP-34124), to perform the boron mixing and natural circulation cooldown test at Palo Verde Unit 1 prior to exceeding 5% power at Palo Verde Unit 2. The staff finds this schedule acceptable.

Auxiliary Pressurizer Spray System

The Palo Verde Nuclear Generating Station was designed without power operated relief valves (PORVs) on the pressurizer. The plant design relies on the APSS as a means of rapidly depressurizing the primary coolant system for plant shutdown and, in the original design basis, for accident mitigation. Since the APSS performs safety-related functions, the applicant asserted that it has been designed to safety-grade standards.

On September 12, 1985, the applicant conducted a loss-of-load test on the Palo Verde Nuclear Generating Station Unit 1 from approximately 55% power. The plant did not perform as expected. The test resulted in an event involving loss of all offsite power to non-essential loads (including the reactor coolant pumps), turbine trip and reactor trip. The reactor and turbine trips were not expected. During the recovery phase of the event, overcooling of the reactor coolant system (RCS) occurred to the extent that the emergency core cooling systems were automatically initiated, followed by the automatic initiation of containment isolation.

The following two sequences occurred during the event that caused the loss of all three charging pumps:

(1) When the safety injection actuation signal (SIAS) occurred, power to certain suction valves for the charging pumps was lost since the motor control center for these valves was classified as non-essential; and, accordingly, was designed to be automatically shed from the safety related electric buses.

(2) Because of a malfunction of the single water level instrument channel for the volume control tank (VCT), automatic control action was lost which would have transferred the suction of the charging pumps from the VCT to other water sources, if power supplies had been available to realign the valves involved. Also, after the containment isolation signal was received, all makeup flow to the VCT was isolated.


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Due to the above sequences, the VCT emptied, the charging pumps became bound on VCT hydrogen cover gas, and the pumps were tripped. This produced a potentially hazardous situation when, to re-establish charging pump flow, the vent lines from the pumps were locally opened by an operator in an attempt to vent the hydrogen gas. However, this attempt was unsuccessful and the charging pumps remained gas bound. After non-class 1E power was restored, water supply from the refueling water tank (RWT) via the boric acid makeup pumps was delivered to the charging pumps, and charging flow to RCP seal injection and reactor coolant system were established. Subsequently, the RCS pressure and inventory reached stable conditions, and the unusual event was terminated.

The applicant's letter of September 18, 1985 (ANPP-33487), discussed the September 12, 1985 event and briefly addressed concerns relating to the APSS. At the conclusion of the September 20, 1985 meeting with the staff, the applicant committed to certain short-term compensatory measures which justified continued operation of the facility while the long-term corrective actions were developed. On October 2, 1985, a letter was issued to the applicant, pursuant to 10 CFR 50.54(f), requiring that the applicant furnish in writing, under oath or affirmation, its plans, program and schedule to bring Palo Verde Unit 1 into conformance with its licensing basis. A request for additional information was enclosed with this letter concerning the design of the APSS relative to safety-grade standards. In response to this staff request, by letters dated October 15, 1985 (ANPP-33713), October 22, 1985 (ANPP-33771), and November 4, 1985 (ANPP-33905), the applicant provided the following:

(1) Reanalyses of the steam generator tube rupture (SGTR) accidents were submitted. In one case, the APSS was assumed to be initiated at two hours following the event and, in another case, the APSS was assumed inoperable and the safety-grade gas vent system from the pressurizer was used for accident mitigation. In both cases, the reanalyses showed that the radiological consequences were within the limits of 10 CFR 100 guidelines.

(2) The applicant proposed four modifications to the Palo Verde design to improve the operator's ability to operate the charging/auxiliary spray system from the control room, to provide an automatic function to reduce the amount of required operator action, and to improve the reliability of control grade level instrumentation on the VCT. The proposed modifications are:

    (a) Provide power to Valves CH-501 and CH-536 from a 1E motor control center (MCC).

    (b) Enhance automatic realignment to the RWT. The modified design would allow automatic realignment to the RWT gravity feed line via Valve CH-536 on lo-lo VCT level when the non-class 1E powered Boric Acid Make-Up Pump flow path is unavailable due to a loss of off-site power.

    (c) Enhance VCT level instrumentation. The modified VCT level instrumentation design would include separate reference legs, one wet and one dry, one to each of the two existing level transmitters, LT-226 and LT-227. This diverse redundancy minimizes the potential for incorrect level indication by eliminating the potential for a partially drained



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wet reference leg going undetected. A signal comparator will be added to the level transmitters, initiating an alarm in the control room when a level difference is indicated. This alarm will alert the operator to possible incorrect indication or malfunction of either transmitter.

(d) Lock open Valves CH-524 and CH-532 to ensure a flow path to the APSS.

The applicant proposed the following schedule for implementation of these modifications:

Unit 1: Following completion of engineering and procurement, currently in process, implementation will be during the first outage of sufficient duration but not later than the completion of the first refueling outage.

Unit 2: Prior to exceeding 5% power.

Unit 3: Prior to fuel load.

The applicant also committed to the following Technical Specification changes:

(1) Include both the pressurizer and the reactor head gas vent flow paths in the LCOs in Section 3.4.10.

(2) Clarify the bases of Section 3.1.2.2 to include Valve CH-501 as part of the required boron injection flow paths.

(3) RWT gravity feedline Valve HV 532 and charging flow path containment isolation Valve CH-524 will be locked to their open position to ensure a flow path to the APSS. Following the system modification of these valves (prior to exceeding 5% power at Palo Verde Unit 2), the Technical Specification will include these requirements in the LCOs.

The staff has reviewed the above submittals and information, and independently verified the results of the SGTR reanalyses. The staff concludes that the radiological consequences of these events, as reanalyzed, are acceptable and that the APSS is not needed for mitigating the design basis SGTR. However, operation of the APSS is assumed for achieving plant cold shutdown in accordance with BTP RSB 5-1. In reviewing the information submitted, the staff determined that the applicant's claim that the charging pumps could be successfully vented needed more definitive substantiation. The staff requested that the applicant address the following:

(1) Provide a description and PI&Ds of the gas vent system from the pressurizer.

(2) Discuss the procedures to be used for venting the charging pumps relative to (a) possible hydrogen explosion and operator's safety while performing the venting process and (b) possible damage to the charging pumps.

(3) Provide a description of the water inventory in the RWT above the upper connection.



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(4) Provide clarification of the equipment qualification applied to Valves CH-239 and CH-240, and the regenerative heat exchanger inside the containment.

In response to the staff request, the applicant provided the following in its letter dated November 25, 1985 (ANPP-34103):

(1) A description and a system diagram showing the gas vent system from the pressurizer to confirm that the system is designed as a safety-grade system.

(2) (a) With respect to the staff concern on the potential for a hydrogen explosion and the operator's safety during gas venting following a possible gas binding of the charging pumps, the applicant stated that the potential for a hydrogen explosion while venting the the charging pump suction piping is eliminated because the vents the operator will utilize are located in close proximity to the HVAC return vents for the area. Therefore, the vented hydrogen will be quickly carried into the HVAC system and dispersed in the high volume air flow. The operator also has the option of using tygon tubing to directly connect the vent valves to the HVAC system. Additionally, the operators have portable detectors available which can warn them of the presence of hydrogen in the work area. The applicant stated that these measures will eliminate the potential for a hydrogen explosion and ensure operator safety during venting process.

(b) With respect to the staff concern on the potential for damage to the charging pumps while running with entrained hydrogen, the applicant stated that this concern was reviewed with the pump vendor and CE. It was determined that operation of the pumps with gas or a gas/water mixture could result in the generation of internal pressure spikes sufficient to crack the pump casing block. Subsequent inspection and performance tests have determined that the Palo Verde Unit 1 pumps were not damaged in this manner during the September 12, 1985 event. In addition, the pump vendor has indicated, and industry experience has verified, that a cracked casing block is not a catastrophic failure and does not cause the pump to fail to operate. Small amounts of leakage would occur but the pump's ability to deliver flow would not be significantly compromised.

(3) A description of the water inventory in the RWT above the upper connection that confirms there is sufficient water supply from the RWT via Valve CH-536 to meet the staff's positions of BTP RSB 5-1.

(4) A confirmation that the charging loop isolation valves (CH-239 and CH-240) are included with the applicable portions of the environmental, mechanical, and seismic qualification programs. These valves are fully qualified to perform their safety related functions. Also, the regenerative heat exchanger meets the requirements of the mechanical equipment and seismic qualification programs. The qualification of the degradable parts of the regenerative heat exchanger is addressed in the mechanical equipment qualification program.



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With respect to items 1, 3 and 4 above, the staff concludes that the applicant has provided responses to resolve the respective issues. However, the information provided to address items 2(a) and 2(b) was not sufficient.

With respect to item 2(a), in the telecon of December 2, 1985, the applicant has confirmed that the charging pump room HVAC systems are not designed to safety-grade standards. Thus, the HVAC system would not be operable for venting hydrogen in the postulated scenario with offsite power unavailable. The staff finds that venting hydrogen gas into unvented spaces is not desirable and required the applicant to take appropriate short-term and long-term measures to vent any accumulation of hydrogen into a HVAC system that would be available, or to provide a venting location (tank, etc.) that would not result in the release of hydrogen or primary coolant to an unmonitored location. In response to the staff request, the applicant committed to the following in its letter dated December 5, 1985 (ANPP-34174):

> The interim operation of the charging system requires procedures for venting of hydrogen from a charging pump(s) which has become gas bound. This venting will not pose a hazard to the operators performing the venting, nor will it result in an uncontrolled or unmonitored radioactive release. The hydrogen gas will not be vented into the charging pump rooms. Procedures will be submitted for staff approval and a demonstration of the venting procedure will be performed prior to Unit 2 initial criticality.
>
> While the proposed enhancements to the VCT level indicator and outlet valve (CH-501) lessen the likelihood of the charging pumps becoming gas bound, a long-term solution to reduce this potential hazard to the pumps will be considered to achieve an appropriate level of system reliability. The long-term solution will be established by an engineering evaluation considering alternative hardware changes necessary to eliminate the need for venting hydrogen from the suction of the charging pumps. The engineering evaluation will include the schedules for procurement and installation of the selected solution. The evaluation and schedules for implementation will be submitted for staff approval by June 30, 1986.

The staff has reviewed the above commitments and concluded that they are acceptable.

With respect to item 2(b), in information provided by the applicant on November 27, 1985, it was stated that cracking of charging pump casing blocks had occurred and been detected in a number of nuclear plants. These pumps were manufactured by ARMCO and Gaulin. The exact cause and mechanism of the initiation of these cracks is unknown. However, the cause is presently under investigation, and is currently attributed to overstressing of the blocks due to piping vibration loads, and to initiating flaws and cracks smaller than detectable by current manufacturing inspection methods (which are in accordance with ASME Code requirements). These cracks occur at high stress concentration areas within the cylinder blocks and, although they appear as hairlines during pump examination, they evidently propagate and enlarge during pump cycling and permit reactor coolant leakage. The industry experience indicates that this leakage is small (estimated at less than two gallons per minute); no catastrophic failures have been recorded; and the pumps continue to operate and deliver required flow.


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The pump cracking is, therefore, not considered an immediate significant safety hazard.

As described previously, on September 12, 1985, PVNGS Unit 1 experienced a hydrogen gas binding event in its charging pumps. Subsequent examination and performance tests indicated that these pumps were not damaged in this event. The existence of hydrogen gas mixtures is a condition which is apparently unique to the CE reactor plants and, although no cracks or leakage have so far been detected at the Palo Verde plant, these pumps have been in operation for too short a period to determine, with reliability, the risk of continued operation.

Neither the industry nor the applicant has provided conclusive documentation that establishes the root cause of the cracking reported in positive displacement charging pump blocks in some C-E operating reactors. Pumps from at least two manufacturers have experienced cracking. In addition, the flow conditions and the piping system configuration apparently influence the potential for cracking. The staff intends to address the issue of the degradation of the charging pumps on a generic basis and will require appropriate remedial action.

The staff has reviewed the applicant's submittals of November 28, 1984 (ANPP-31264), and November 25, 1985 (ANPP-43103), in which the applicant proposed specific actions after the issuance of an operating license for Palo Verde Unit 2. Based on the information provided, and the fact that the charging pumps are safety-grade components and are relied on for reaching cold shutdown following the design basis loss of offsite power event, the staff believes the following measures are necessary in order to reduce the probability of damage to the charging pumps:

(1) Perform baseline nondestructive examination using surface examination techniques, e.g., liquid penetrant, magnetic particle or ultrasonic sound examination of all charging pump blocks at locations where cracks were detected on other pumps.

(2) Perform weekly visual examinations of each pump with appropriate aids suitable for the detection of cracking.

(3) Declare the specific pump inoperable upon confirmation of cracking.

(4) Perform a demonstration of the hydrogen gas venting process at the charging pumps prior to exceeding 5 % power at Palo Verde Unit 2.

(5) Provide a detailed evaluation of the charging pump operability when subjected to the various phenomena associated with hydrogen gas binding. This evaluation will be completed prior to the end of the first refueling outage of Palo Verde Unit 1.

In letters dated November 29, 1985 (ANPP-34127), and December 5, 1985 (ANPP-34174), the applicant committed to perform action items 2, 4 and 5 described above. For item 1, the applicant has committed to perform an external surface examination on the Unit 2 charging pumps utilizing the appropriate NDE technique. This examination will be completed prior to Unit 2 initially exceeding


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5% power. For item 3, the applicant has committed that it will declare the charging pump inoperable within 72 hours of the apparent thru wall crack discovery. The staff concludes that the applicant's commitments are acceptable.

In summary, the staff concludes that the enhancements, technical specifications and schedules described above will improve the reliability of the APSS and charging pump water supply systems and provide reasonable assurance that the APSS will be available for plant cold shutdown per BTP RSB 5-1. These measures are, therefore, acceptable.


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# 6 ENGINEERED SAFETY FEATURES

## 6.3 Emergency Core Cooling System

By letter dated October 3, 1985, the applicant has re-analyzed the limiting large break LOCA to demonstrate that PVNGC Units 1, 2 and 3 are in compliance with 10 CFR 50.46. The NRC requested in letter dated August 30, 1985, that this analysis be performed to account for an error in the treatment of the axial power shape in the Combustion Engineering Evaluation Model. The revised analysis uses a 1.52 axial peak located above the core mid-plane with a corresponding increase in the radial peaking factor to maintain the hot rod power peak at 14.0 Kw/ft, the current PVNGS Technical Specification limit. In addition to the more conservative axial power shape, the PVNGS plant specific containment parameters (volume and passive heat sinks) and the Safety Injection Pump flow performance data were used in the re-analysis.

The changes for the PVNGS containment parameters and the calculation of the large break LOCA containment back-pressure may be found in Section 6.2.1.5 of the FSAR (Amendment 14). These data have been accepted by the staff for reference in this analysis.

The minimum performance characteristics of the PVNGS Safety Injection System (SIS) pump flows, as discussed in the Unit 1 low power license (NPF-34) Condition 2.C.(21), were used for this analysis. The axial power peak value of 1.52 is more restrictive than would be allowed based on the current axial shape index Technical Specification limit of -0.28. Therefore, the results of this re-analysis are conservative, and the current Technical Specification regarding power shapes and linear heat generator rates are acceptable. The current approved Combustion Engineering Large Break LOCA Evaluation Model was used to perform the reanalysis.

The plant specific PVNGS large break LOCA re-analysis results in a peak cladding temperature of 2091°F. The peak local oxidation is 9.02%, and the core-wide oxidation is less than 0.799%. These results are all acceptable since they are within the following 10 CFR 50.46 Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Reactors:

(1) The peak cladding temperature is less than 2200°F.

(2) The peak local oxidation is less than 17% of the cladding thickness before oxidation.

(3) The core-wide oxidation is less than 1% of the hypothetical maximum.

## 6.6 Inservice Inspection of Class 2 and 3 Components

This evaluation provides additional conclusions to supplement information in this section of the SER and Supplements 1 and 3, which address the definition of examination requirements and the evaluation of compliance with 10 CFR 50.55a(g).



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6.6.1 Evaluation of Compliance for Palo Verde Unit 1 With 10 CFR 50.55a(g).

The SER states that the initial inservice inspection (ISI) program had not been submitted for review. In a letter dated August 26, 1985 the applicant submitted the Unit 1 ISI program manual for review and approval. The staff discussed this document in Section 5.2.4.1 of this supplement.

6.6.2 Evaluation of Compliance for Palo Verde Units 2 and 3 With 10 CFR 50.55a(g)

SSER 3 states that the applicant was considering the updating of the preservice inspection (PSI) program to meet later requirements of ASME Code Section XI. In a letter dated February 5, 1985, the applicant provided a comparison of the PSI programs for all units and information specific to Units 2 and 3. The PSI programs for PVNGS 1-3 are similar to the programs for other CE plants that have been docketed and consist of shop and field examinations. The principal components, such as the steam generators, were examined in the fabrication shop. Certain welds in these principal components and the welds in the balance of the plant components, subject to the preservice examination sample, were conducted during the field examinations. A volumetric examination of at least 10% of the welds over 4 inches in diameter in the ECC, RHR and CHR systems is included in the field PSI programs for PVGNS 1-3.

In SSER 1 the staff found acceptable the selection of the ASME Code Class 2 and 3 of the welds subject to examination in the Unit 1 PSI program. The regulation permits all components (including supports) to meet the requirements set forth in subsequent ASME Code editions and addenda referenced in 10 CFR 50.55a(b). The staff has determined the applicant used acceptable methods to update the PSI programs for Units 2 and 3. Therefore, the staff concludes that the sample welds of ASME Class 2 and 3 components subject to preservice examination in Units 2 and 3 is acceptable.

The applicant requested relief from certain Code preservice requirements that the applicant determined to be impractical to perform on Unit 2 and provided a supporting justification in letters dated February 5, 1985 and July 5, 1985. The technical issues are similar to those identified on Unit 1 and addressed in SSER 3. The staff evaluated the ASME Code-required examinations that the applicant determined to be impractical and the staff determined that the applicant has demonstrated that either (1) the proposed alternatives would provide an acceptable level of quality and safety or (2) compliance with the requirements would result in hardships or unusual difficulties without a compensating increase in the level of quality and safety. Based on the review of the applicant's submittals and the granting of relief from these preservice examination requirements, the staff concludes that the PSI program for PVNGS Unit 2 is acceptable and in compliance with 10 CFR 50.55a(g)(3). The detailed evaluation supporting this conclusion is provided in Appendix L to this report. The staff will evaluate requests for relief from impractical examinations on Unit 3 after detailed plant-specific information is provided by the applicant. The staff will report its conclusions in a future supplement to the SER.



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## 11 RADIOACTIVE WASTE MANAGEMENT

### 11.3 Process and Effluent Radiological Monitoring Systems

This issue is discussed in Sections 11.3 and 12.3.4 of the SER.

The applicant, in its letter dated November 29, 1985, has provided information and justification for interim operation of their radiation monitoring system for Unit 2. The letter proposed changes which would modify appropriate sections of Section 11 and 12 of their FSAR. Specifically, the applicant has identified each radiation monitor: as (1) required by Technical Specification, (2) required by Regulatory Guide 1.97; or (3) a non-technical specification monitor. Each type of monitor would be made operable by a specific due date.

The applicant has committed the following schedule for making all radiation monitors operable: (1) technical specification monitors would be made operable prior to the mode for which they are required without entering a technical specification action statement; (2) Regulatory Guide 1.97 monitors would be made operable per the applicant's previous commitments made in letter dated November 8, 1985; (3) non-technical specification monitors would be made operable per the commitment made in letter dated November 29, 1985. Item No. 8 to Attachment 1 of the Unit 2 low power license has been added to address these commitments. The applicant has also provided their committed compensatory measures and justification for any monitor not being operable prior to initial criticality.

The staff has evaluated the applicant's proposal using the criteria stated in NUREG-0800, (SRP) Sections 11 and 12 and finds the request acceptable.


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# 13 CONDUCT OF OPERATIONS

## 13.1 Organizational Structure of Applicant

### 13.1.2 Operating Organization

#### 13.1.2.3 Operating Shift Crews

Palo Verde Unit 1 received its low power license on December 31, 1984, and its full power license on June 1, 1985. The plant entered operating Mode 3 on April 29, 1985. The staff has agreed with the applicant that time of entry into Mode 3 operation is a suitable point to start crediting time for the purpose of establishing the "hot operating experience" of the plant operators.

The applicant presently forecasts readiness for a low power license for Unit 2 in December 1985. By November 1, 1985, the operators who have participated in the entire startup program on Unit 1 will have accumulated credit for six months of hot operating experience on that unit. It is the applicant's intention that these Unit 1 operators will be dual-licensed on Unit 2, such that they can participate in or provide backup support for the Unit 2 startup.

By letter dated August 30, 1985, the applicant provided detailed plans for senior operator staffing for Unit 2 startup. By November 1, 1985, 18 senior operators will have accumulated at least 6 months of hot operating experience either on Palo Verde Unit 1 or a similar type plant, including at least 6 weeks above 20% power. The staff has reviewed the information provided by the applicant and agrees that a sufficient number of experienced senior operators are available to meet the operating experience guidelines of Generic Letter 84-16 for Unit 2 startup while still retaining a minimum base of operating experience on each of the Unit 1 operating shifts.

The staff concludes that the applicant's plans for providing hot operating experience to each of the Unit 2 operating shifts are acceptable. The operating license for Unit 2 will be conditioned to ensure that this operating experience is provided.

## 13.6 Physical Security

Section 13.6 of SSERs 5 and 7 discussed the resolution of physical security for Unit 1 only. The following staff evaluation of Unit 2's site security plan is based on supplemental information received from the applicant dated September 20, 1985.

### Introduction

The Arizona Public Service Company has filed with the Nuclear Regulatory Commission for the Palo Verde Nuclear Generating Station Unit 2 the following security plans:



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"Palo Verde Nuclear Generating Station Physical Security Plan", (the Contingency Plan is contained in Chapter 8) and the "Palo Verde Nuclear Generating Station Guard Training and Qualification Plan."

This Safety Evaluation Report (SER) summarizes how the applicant has provided for meeting the requirements of 10 CFR Part 73. The SER is composed of a basic analysis that is available for public review, a protected Appendix, and a protected response force size worksheet.

Based on a review of the subject documents and visits to the site, the staff has concluded that the protection provided by the Arizona Public Service Company against radiological sabotage at the Palo Verde Nuclear Generating Station Unit 2 meets the requirements of 10 CFR Part 73. Accordingly, the protection provided will ensure that the health and safety of the public will not be endangered.

## Physical Security Organization

To satisfy the requirements of 10 CFR 73.55(b) Arizona Public Service Company has provided a physical security organization that includes a Security Shift Supervisor who is onsite at all times with the authority to direct the physical protection activities. To implement the commitments made in the physical security, guard training and qualification plan, and the safeguards contingency plan, written security procedures specifying the duties of the security organization members are available for inspection. The training program and critical security tasks and duties for the security organization personnel are defined in the "Palo Verde Nuclear Generating Station Guard Training and Qualification Plan" which meets the requirements of 10 CFR Part 73, Appendix B for the training, equipping and qualification of the security organization members. The physical security plan and the training program provide commitments that preclude the assignment of any individual to a security related duty or task prior to the individual being trained, equipped and qualified to perform the assigned duty in accordance with the approved guard training and qualification plan.

## Physical Barriers

In meeting the requirements of 10 CFR 73.55(c) the licensee has provided a protected area barrier which meets the definition of 10 CFR 73.2(f)(1). An isolation zone, to permit observation of activities along the barrier of at least 20 feet is provided on both sides of the barrier with the exception of the locations listed in the Appendix. The staff reviewed those locations and determined that the security measures in place are satisfactory and continue to meet the requirements of 10 CFR 73.55(c).

Illumination of 0.2 foot-candle is maintained for the isolation zones, protected area barrier, and external portions of the protected area. In areas where illumination of 0.2 foot-candle cannot be maintained, special procedures are applied as described in the Appendix.

Patrols of the protected area are performed at random intervals to detect the presence of unauthorized persons, vehicles, and materials.



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Identification of Vital Areas

The Appendix contains a discussion of the applicant's vital area program and identifies those areas and items of equipment determined to be vital for protection purposes. Vital equipment is located within vital areas which are located within the protected area and which require passage through at least two barriers, as defined in 10 CFR 73.2(f)(1) and (2) to gain access to vital equipment.

Vital area barriers are separated from the protected area barrier. The control room and central alarm station are provided with bullet-resistant walls, doors, ceilings, floors, and windows. Based on these findings and the analysis set forth in the Appendix, the staff has concluded that the applicant's program for identification and protection of vital equipment satisfies the regulatory intent. However, this program is subject to onsite validation by the staff in the future, and to subsequent changes if found to be necessary.

Access Requirements

In accordance with 10 CFR 73.55(d), all points of personnel and vehicle access to the protected area are controlled. The individual responsible for controlling the final point of access into the protected area is located in a bullet-resistant structure. As part of the access control program, vehicles (except under emergency conditions), personnel, packages, and materials entering the protected area are searched for explosives, firearms and incendiary devices by electronic search equipment and/or physical search.

Vehicles admitted to the protected area, except licensee designated vehicles, are controlled by escorts. Licensee designated vehicles are limited to onsite station functions and remain in the protected area except for operational maintenance, repair, security, and emergency purposes. Positive control over these vehicles is maintained by personnel authorized to use the vehicles or by the escort personnel.

A picture badge/key card system, utilizing encoded information, identifies individuals that are authorized unescorted access to protected and vital areas and is used to control access to these areas. Individuals not authorized unescorted access are issued nonpicture badges that indicate an escort is required. Access authorizations are limited to those individuals who have a need for access to perform their duties.

Unoccupied vital areas are locked and alarmed. Access to the reactor containment is positively controlled to assure that only authorized individuals are permitted to enter. In addition, all doors and personnel/equipment hatches into the reactor containment are locked and alarmed. Keys, locks, combinations, and related equipment are changed on an annual basis. In addition, when an individual's access authorization has been terminated due to the lack of reliability or trustworthiness, or for poor work performance, the keys, locks, combinations, and related equipment to which that person had access are changed.

Detection Aids

In satisfying the requirements of 10 CFR 73.55(e), the licensee has installed intrusion detection systems at the protected area barrier, at entrances to


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vital areas, and at all emergency exits. Alarms from the intrusion detection system annunciate within the continuously manned central alarm station located in the protected area and within a secondary alarm station also located in the protected area. In addition, the central alarm station is constructed so that the walls, floors, ceilings, doors, and windows are bullet-resistant. The alarm stations are located and designed in such a manner so that a single act cannot interdict the capability of calling for assistance or responding to alarms. The central alarm station contains no other functions or duties that would interfere with its alarm response function.

The intrusion detection system transmission lines and associated alarm annunciation hardware are line-supervised and tamper-indicating. Alarm annunciators indicate the type of alarm and its location when activated. An automatic indication of when the alarm system is on standby power is provided in the central alarm station.

Communications

As required in 10 CFR 73.55(f), the licensee has provided for the capability of continuous communications between the central and secondary alarm station operators, guards, watchmen, and armed response personnel through the use of a conventional telephone system and a security radio system. In addition, direct communication with the local law enforcement authorities is maintained through the use of a conventional telephone system and a two-way FM radio link.

All nonportable communication links, except the conventional telephone system, are provided with an uninterruptible emergency power source backed up by diesel generators.

Test and Maintenance Requirements

In meeting the requirements of 10 CFR 73.55(g), the licensee has established a program for the testing and maintenance of all intrusion alarms, emergency alarms, communication equipment, physical barriers, and other security related devices or equipment. Equipment or devices that do not meet the design performance criteria or have failed to otherwise operate will be compensated for by appropriate compensatory measures as defined in the "Palo Verde Nuclear Generating Station Physical Security Plan" and in site procedures. The compensatory measures defined in these plans will assure that the effectiveness of the security system is not reduced by failures or other contingencies affecting the operation of the security related equipment or structures.

Intrusion detection systems are tested for proper performance at the beginning and end of any period that they are used for security. Such testing will be conducted at least once every seven days. Communication systems for onsite communications are tested at the beginning of each security shift. Offsite communications are tested at least once each day.

Audits of the security program are conducted once every 12 months by personnel independent of site security management and supervision. The audits, focusing on the effectiveness of the physical protection provided by the onsite security organization in implementing the approved security program plans, include, but are not limited to: a review of the security procedures and practices; system


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testing and maintenance programs; and local law enforcement assistance agreements. Arizona Public Service quality assurance and corporate security personnel prepare a report documenting audit findings and recommendations.

Response Recommendations

In meeting the requirements of 10 CFR 73.55(h), the licensee has provided for armed responders immediately available for response duties on all shifts consistent with the requirements of the regulations (see Appendix). Considerations used in support of this number are attached. In addition, liaison with local law enforcement authorities to provide additional response support in the event of security events has been established and documented.

The applicant's safeguards contingency plan for dealing with thefts, threats, and radiological sabotage events satisfies the requirements of 10 CFR Part 73, Appendix C. The plan identifies appropriate security events which could initiate a radiological sabotage event and identifies the applicant's preplanning, response resources, safeguards contingency participants and coordination activities for each identified event. Through this plan, upon the detection of abnormal presence or activities within the protected or vital areas, response activities using the available resources would be initiated. The response activities and objectives include the neutralization of the existing threat by requiring the response force members to interpose themselves between the adversary and their objective, instructions to use force commensurate with that used by the adversary, and authority to request sufficient assistance from the local law enforcement authorities to maintain control over the situation.

Employee Screening Program

In meeting the requirements of 10 CFR 73.55(a) to protect against the design basis threat as stated in 10 CFR 73.1(a)(1)(ii), Arizona Public Service Company has provided for an employee screening program. Personnel who successfully complete the employee screening program or its equivalent may be granted unescorted access to protected and vital areas at the Palo Verde site. All other personnel requiring access to the site are escorted by persons authorized and trained for escort duties and who have successfully completed the employee screening program. The employee screening program is based upon accepted industry standards and includes a background investigation, psychological evaluation, and a continuing observation program.

The plan also provides for a "grandfather clause" exclusion which allow recognition of a certain period of trustworthy service with the utility or contractor as being equivalent to the overall employee screening program. The staff has reviewed the applicant's screening program against the accepted industry standards (ANSI N18.17 1973) and has determined that the Arizona Public Service Company program is acceptable.



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22 TMI-2 REQUIREMENTS

22.2 Evaluation of TMI Requirements

II.B.3 Post-Accident Sampling Capability

In the SER, the staff concluded that the applicant's proposed Post-Accident Sampling System (PASS) met all the eleven criteria of Item II.B.3 in NUREG-0737 and was acceptable.

By letters dated August 19, September 26 and October 24, 1985, the applicant proposed modifications to the post-accident sampling system.

Evaluation

Criterion (1)

> The licensee shall have the capability to promptly obtain reactor coolant samples and containment atmosphere samples. The combined time allotted for sampling and analysis should be 3 hours or less fr e time a decision is made to take a sample.

The Post-Accident Sampling System will use grab sample method gy as its primary means of sampling reactor coolant and containment atmosphere. The PASS can be operated from the hot lab to obtain reactor coolant and containment atmosphere samples and analyze any required sample within the required time span of 3 hours from the time a decision is made to take a sample. A manually loaded Class IE power source can be available within 30 minutes of an accident that assumes a loss of offsite power to allow the acquisition of reactor coolant and containment atmosphere samples.

The staff determined that these provisions meet Criterion (1) and are, therefore, acceptable.

Criterion (2)

> The licensee shall establish an onsite radiological and chemical analysis capability to provide, within the 3-hour time frame established above, quantification of the following:
>
> (a) certain radionuclides in the reactor coolant and containment atmosphere that may be indicators of the degree of core damage (e.g., noble gases, iodines and cesiums, and nonvolatile isotopes);
>
> (b) hydrogen levels in the containment atmosphere;
>
> (c) dissolved gases (e.g., $H_2$), chloride (time allotted for analysis subject to discussion below), and boron concentration of liquids;



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(d) alternatively, have inline monitoring capabilities to perform all or part of the above analyses.

Liquid and gaseous grab samples will be obtained from the reactor vessel hot leg, letdown line (cold leg), containment sump, and containment atmosphere for analyses of radioisotopes, pH, boron, chloride, dissolved gases, hydrogen, and oxygen.

The procedure to be utilized to estimate the degree of core damage was developed from the "Development of Comprehensive Procedure Guidelines for Core Damage Assessment", Combustion Engineering Owners Group Task 467, dated July 1983. The procedure uses radioisotopic analysis data and takes into consideration other physical parameters, such as local core exit thermocouple temperatures, core coolant conditions, hydrogen concentrations, and area radiation levels. The staff finds that these provisions meet Criterion (2) and are, therefore, acceptable.

Criterion (3)

Reactor coolant and containment atmosphere sampling during post-accident conditions shall not require an isolated auxiliary system (e.g., the letdown system, reactor water cleanup system (RWCSU)) to be placed in operation in order to use the sampling system.

Operation of the PASS to obtain reactor coolant and containment atmosphere samples does not require an isolated auxiliary system to be placed into service. All equipment necessary to obtain reactor coolant and containment atmosphere samples has been verified to be operable in the environment it would experience under accident conditions.

The staff finds that these provisions meet Criterion (3) and are, therefore, acceptable.

Criterion (4)

Pressurized reactor coolant samples are not required if the licensee can quantify the amount of dissolved gases with unpressurized reactor coolant samples. The measurement of either total dissolved gases or $H_2$ gas in reactor coolant samples is considered adequate. Measuring the $O_2$ concentration is recommended, but is not mandatory.

The PASS has the capability to analyze depressurized off gas from reactor coolant grab samples for total dissolved gases and dissolved hydrogen, in the concentration ranges of 11-2000 cc/kg and 10-2000 cc/kg, respectively.

The staff determined that these provisions meet Criterion (4) and are, therefore, acceptable.

Criterion (5)

The time for a chloride analysis to be performed is dependent upon two factors: (a) if the plant's coolant water is seawater or brackish water and (b) if there is only a single barrier between primary containment systems and the cooling water. Under both of the above conditions the



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licensee shall provide for a chloride analysis within 24 hours of the sample being taken. For all other cases, the licensee shall provide for the analysis to be completed within 4 days. The chloride analysis does not have to be done onsite.

Reactor coolant samples can be analyzed by ion chromatography for chloride ions within 96 hours (4 days), with an analytical range of 0.02-20 ppm and an accuracy of ± 25%.

The staff determined that these provisions meet Criterion (5) and are, therefore, acceptable.

Criterion (6)

The design basis for plant equipment for reactor coolant and containment atmosphere sampling and analysis must assume that it is possible to obtain and analyze a sample without radiation exposures to any individual exceeding the criteria of GDC 19 (Appendxix A, 10 CFR Part 50) (i.e., 5 rem whole body, 75 rem extremities). (Note that the design and operational review criterion was changed from the operational limits of 10 CFR Part 20 (NUREG-0578) to the GDC 19 criterion (October 30, 1979 letter from H. R. Denton to all licensees).)

Plant personnel can obtain and analyze post-accident grab samples without radiation exposures to any individual in excess of 5 rem whole body and 75 rem to the extremities (General Design Criterion 19).

The staff finds that these provisions meet Criterion (6) and are, therefore, acceptable.

Criterion (7)

The analysis of primary coolant samples for boron is required for PWRs. (Note that Revision 2 of Regulatory Guide 1.97, when issued, will likely specify the need for primary coolant boron analysis capability at BWR plants).

Reactor coolant samples can be analyzed for boron. The analytical range is 100 to 6000 ppm, and the accuracy is ± 50 ppm below 1000 ppm and ± 5% above 1000 ppm boron.

The staff determined that these provisions meet Criterion (7) and are, therefore, acceptable.

Criterion (8)

If inline monitoring is used for any sampling and analytical capability specified herein, the licensee shall provide backup sampling through grab samples, and shall demonstrate the capability of analyzing the samples. Established planning for analysis at offsite facilities is acceptable. Equipment provided for backup sampling shall be capable of providing at least one sample per day for 7 days following onset of the



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accident and at least one sample per week until the accident condition no longer exists.

The primary means for sampling the reactor coolant and containment atmosphere is a grab sample type PASS. Grab samples are analyzed in an appropriate laboratory facility.

The staff finds that these provisions meet Criterion (8) and are, therefore, acceptable.

Criterion (9)

The licensee's radiological and chemical sample analysis capability shall include provisions to:

(a) Identify and quantify the isotopes of the nuclide categories discussed above to levels corresponding to the source terms given in Regulatory Guides 1.3 or 1.4 and 1.7. Where necessary and practicable, the ability to dilute samples to provide capability for measurement and reduction of personnel exposure should be provided. Sensitivity of onsite liquid sample analysis capability should be such as to permit measurement of nuclide concentration in the range from approximately 1 μCi/g to 10 Ci/g.

(b) Restrict background levels of radiation in the radiological and chemical analysis facility from sources such that the sample analysis will provide results with an acceptably small error (approximately a factor of 2). This can be accomplished through the use of sufficient shielding around samples and outside sources, and by the use of ventilation system design which will control the presence of airborne radioactivity.

Isotopic analysis can be performed to identify and quantify the isotopes of the nuclide categories corresponding to the source terms given in Regulatory Guides 1.4 and 1.7. Provisions are included to measure a wide range of isotopes for both gases and liquids from 10 μCi/ml to 10 Ci/ml Sample dilution is available for grab sample analysis.

If background levels of radiation are too high in the sample analysis area to permit the analysis of the grab samples obtained, the sample can be transported to an unaffected PVNGS unit laboratory for analysis.

Background levels of radiation in the sample analysis area of the hot lab are kept ALARA, and the analytical results will have an error approximately a factor of 2.

Grab samples can be taken with a shielded sample syringe and transported with a lead PIG.

The hot lab is provided with a ventilation system which will control the presence of airborne radioactivity.

The staff finds that these provisions meet Criterion (9) and are, therefore, acceptable.



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Criterion (10)

Accuracy, range, and sensitivity shall be adequate to provide pertinent data to the operator in order to describe radiological and chemical status of the reactor coolant systems.

The PASS has the analytical ranges and accuracies that are consistent with the recommendations of Regulatory Guide 1.97, Rev. 3, and the clarification of NUREG-0737, Item II.B.3, Post-Accident Sampling Capability. The analytical methods and instrumentation were selected for their ability to operate in the post-accident sampling environment. Equipment used in post-accident sampling and analyses will be calibrated or tested at least every six months. Retraining of operators for post-accident sampling is scheduled at a frequency of once every six months. The staff finds that these provisions meet Criterion (10) and are, therefore, acceptable.

Criterion (11)

In the design of the post-accident sampling and analysis capability, consideration should be given to the following items:

(a) Provisions for purging sample lines, for reducing plateout in sample lines, for minimizing sample loss or distortion, for preventing blockage of sample lines by loose material in the RCS or containment, for appropriate disposal of the samples, and for flow restrictions to limit reactor coolant loss from a rupture of the sample line. The post-accident reactor coolant and containment atmosphere samples should be representative of the reactor coolant in the core area and the containment atmosphere following a transient or accident. The sample lines should be as short as possible to minimize the volume of fluid to be taken from containment. The residues of sample collection should be returned to containment or to a closed system.

(b) The ventilation exhaust from the sampling station should be filtered with charcoal adsorbers and high-efficiency particulate air (HEPA) filters.

Provisions are made to purge sample lines, for reducing plate-out,to insure proper mixing, for minimizing leakage, preventing blockage, to back-flush, to blowdown, for appropriate sample disposal, minimizing crud traps, and for passive flow restrictions.

Containment gaseous sample lines for post-accident sampling are heat traced to reduce plate-out of iodines and particulates. All post-accident liquid sample lines are filtered to prevent blockage by loose material in the reactor coolant system or containment sump samples.

The sample source penetrations are located such that reactor coolant and containment atmosphere samples are representative of core area and containment conditions.

Samples are returned to the containment or the reactor drain tank. The ventilation exhaust from the PASS sampling station is filtered through HEPA and charcoal filters located in the Auxiliary Building.



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The staff determined that these provisions meet Criterion (11) and are, therefore, acceptable.

Conclusion

On the basis of the above evaluation, the staff concludes that the Post-Accident Sampling System with the proposed modifications meets all of the 11 criteria in Item II.B.3 of NUREG-0737 and is, therefore, acceptable. However, as stated in the applicant's letter of September 26, 1985, the PASS will be operable prior to exceeding 5% of full power. An appropriate license condition has been included in the Unit 2 low power license.



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APPENDIX A

CONTINUATION OF CHRONOLOGY OF RADIOLOGICAL REVIEW

| | |
|---|---|
| May 21, 1985 | Letter to licensee regarding petition on spray pond piping corrosion and its effects |
| May 23, 1985 | Letter from licensee forwarding emergency plan implementing procedure update |
| May 24, 1985 | Letter from licensee concerning compliance with the Emergency Planning Statement of Policy |
| May 29, 1985 | Letter from licensee transmitting inadequate core cooling instrumentation system implementation report |
| June 1, 1985 | Issuance of Facility Operating License No. NPF-41 authorizing 100% power and issuance of Supplement 8 to Safety Evaluation Report |
| June 5, 1985 | Letter from licensee providing clarification to May 24 letter |
| June 18, 1985 | Letter to licensee advising of acceptability of post-accident monitoring instrumentation (Regulatory Guide 1.97) |
| June 18, 1985 | Letter to applicant transmitting request for additional information on safety parameter display system |
| June 21, 1985 | Letter from licensee transmitting response to Generic Letter 85-02, Staff Recommended Actions Stemming from NRC Integrated Program for the Resolution of Unresolved Safety Issues Regrading Steam Generator Tube Integrity |
| June 21, 1985 | Letter to licensee concerning draft technical specifications for Unit 2 |
| June 24, 1985 | Site audit of Unit 1 core protection calculator |
| June 25, 1985 | Site visit to audit inadequate core cooling instrumentation for implementation review |


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| June 27, 1985 | Meeting with licensee to discuss concept of temperature dependent shutdown margin |
| June 27, 1985 | Letter from licensee regarding license condition on post accident dose assessment |
| June 27, 1985 | Meeting with licensee regarding natural circulation tests |
| June 28, 1985 | Generic Letter 85-11- Completion of Phase II of "Control of Heavy Loads at Nuclear Power Plants" |
| June 28, 1985 | Generic Letter 85-12 - Implementation of TMI Action Item II.K.3.5, "Automatic Trip of Reactor Coolant Pumps" |
| June 28, 1985 | Letter from licensee regarding relief requests from requirements of ASME Boiler and Pressure Vessel Code, Section XI, per 10 CFR 50.55(g)(5)(iii) |
| July 3, 1985 | Letter to licensee transmitting safety evaluation concerning its response to Generic Letter 83-28, Items 1.1 and 1.2 |
| July 5, 1985 | Letter from licensee advising of change to its commitment with respect to painted lines on top of secondary shield walls |
| July 5, 1985 | Letter from licensee transmitting technical description associated with Relief Request No. 1 of Appendix B for Unit 2 (pre-service examination program) |
| July 10, 1985 | Meeting with CE and utilities to discuss recent re-evaluation of ECCS analysis |
| July 10, 1985 | Letter from licensee transmitting schedule for revising plant specific technical guidelines (Procedures Generation Package) and emergency operating procedures |
| July 12, 1985 | Letter from licensee regarding potential LOCA analysis non-conservatism |
| July 12, 1985 | Letter from licensee transmitting marked up pages of Unit 2 proposed technical specifications |
| July 16, 1985 | Letter from licensee requesting schedular exemption to GDC 4 for Units 2 and 3 |


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| July 16, 1985 | Letter from licensee concerning resolution of TMI Action Plan Item II.K.3.30 |
| July 19, 1985 | Letter to licensee requesting information on post-LOCA status of reactor coolant pumps seal cooling |
| July 19, 1985 | Letter from licensee advising of voluntary reduction of linear heat rate limit |
| July 26, 1985 | Letter from licensee forwarding emergency plan implementing procedures |
| July 30, 1985 | Letter to licensee regarding schedule for completion of licensing review for Unit 2 |
| August 1, 1985 | Generic Letter 85-14 - Commercial Storage and Power Reactor Sites of Low Level Radwaste Not Generated by Utility |
| August 2, 1985 | Letter from applicant providing additional information for draft technical specifications for Unit 2 |
| August 5, 1985 | Generic Letter 85-13 - Transmittal of NUREG-1154 Regarding Davis-Besse Loss of Main and Auxiliary Feedwater Event |
| August 5, 1985 | Issuance of Amendment No. 1 to NPF-41 confirming emergency authorization granted July 12 to permit, on a one time basis only during the power ascension program, an additional 72 hours in hot standby before proceeding to cold shutdown |
| August 6, 1985 | Generic Letter 85-15 - Information Concerning Deadlines for Compliance with 10 CFR 50.59, "Environmental Qualification of Electric Equipment Important to Safety of Nuclear Power Plants" |
| August 12, 1985 | Letter from licensee advising that information in August 2, 1984 letter on plant protection system setpoint methodology is applicable for Units 2 and 3 |
| August 12, 1985 | Letter from licensee regarding guide tube wear surveillance |
| August 14, 1985 | Letter to licensee forwarding proof and review copy of technical specifications for Unit 2 |



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| August 15, 1985 | Letter from licensee transmitting Monthly Operating Report for July |
| August 19, 1985 | Letter from licensee transmitting additional information on Safety Parameter Display System |
| August 19, 1985 | Letter from licensee transmitting information on fuel load date, Technical Specifications, remaining SER issues and startup testing |
| August 23, 1985 | Letter from licensee transmitting additional information on revised Chapter 15 analyses |
| August 23, 1985 | Generic Letter 85-16 - High Boron Concentrations |
| August 23, 1985 | Generic Letter 85-17 - Availability of Supplement 2 and 3 to NUREG-0933, "Prioritization of Generic Safety Issues" |
| August 27, 1985 | Letter to licensee advising that it has met requirements for TMI Action Plan Item II.K.3.30, Small Break LOCA Analysis and Item II.K.3.31, Plant Analysis |
| August 28, 1985 | Issuance of Bi-weekly Sholly notice regarding one-time exception to tech specs involving reactor coolant system pumps and atmospheric dump valves, to allow performance of Natural Circulation Cooldown Test |
| August 29, 1985 | Letter from licensee transmitting FSAR changes for Units 2 and 3 for review - changes relate to Chapter 1 of FSAR |
| August 29, 1985 | Letter from licensee transmitting FSAR changes for Units 2 and 3 for review - changes relate to Chapter 6 of FSAR |
| August 29, 1985 | Meeting with licensee to discuss scope of small break LOCA analysis |
| August 30, 1985 | Letter from licensee forwarding information on shift staffing |
| August 30, 1985 | Letter to licensee regarding large break LOCA evaluation model for Units 2 and 3 |
| August 30, 1985 | Letter form licensee transmitting "10 CFR 50, Appendix R Safe Shutdown Evaluation, Outside Control Room Fire Spurious Actuation Study" |



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| August 30, 1985 | Letter from Combustion Engineering transmitting "CPC/CEAC Software Modifications for CPC Improvement Program," Revision 0 (proprietary and nonproprietary versions) |
| August 30, 1985 | Letter to licensee advising of approval of ICCI system implementation |
| August 30, 1985 | Letter from licensee transmitting supplemental Detailed Control Room Design Review Summary Report |
| August 30, 1985 | Letter from licensee transmitting FSAR changes for Units 2 and 3 for review - changes relate to Chapters 3, 9, 10, and 11 of FSAR |
| August 30, 1985 | Letter from licensee transmitting FSAR changes for Units 2 and 3 for review - changes relate to Chapters 3, 9, and 13 of FSAR |
| August 30, 1985 | Letter from licensee transmitting FSAR changes for Units 2 and 3 for review - changes relate to various sections |
| August 30, 1985 | Letter from licensee transmitting FSAR changes for Units 2 and 3 for review - changes relate to Chapter 13 of FSAR |
| August 30, 1985 | Letter from licensee transmitting FSAR changes for Units 2 and 3 for review - changes relate to Chapters 9 and 10 |
| August 30, 1985 | Letter from licensee transmitting information on pressurized thermal shock |
| August 30, 1985 | Letter from licensee transmitting FSAR changes for Units 2 and 3 for review - changes relate to testing, Chapters 1 and 14 |
| August 30, 1985 | Letter from licensee transmitting FSAR changes for Units 2 and 3 for review - changes relate to Chapters 14 and 17 |
| August 30, 1985 | Letter from licensee transmitting FSAR changes for Units 2 and 3 for review - changes relate to Chapter 17 |
| September 5, 1985 | Letter from CE transmitting P & NP Report on Core Operating Limit Super. System |
| September 9, 1985 | Form transmittal of August 28, 1985 Sholly notice to licensee |



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|---|---|
| September 10, 1985 | Letter from licensee transmitting information on post-LOCA status of reactor coolant pumps seal cooling |
| September 13, 1985 | Letter to licensee advising that intent of criteria of NUREG-0612, "Control of Heavy Loads," is satisfied |
| September 18, 1985 | letter from licensee requesting that storage of Unit 3 neutron startup sources in Unit 2 fuel building be included in Unit 2 operating license |
| September 19, 1985 | Letter from licensee regarding reexamination of diesel generator cooling and essential cooling water systems at Unit 2 |
| September 20, 1985 | Letter from licensee transmitting Revision 8 of security plan |
| September 23, 1985 | Letter to licensee regarding technical specification schedule for Unit 2 |
| September 26, 1985 | Letter from licensee regarding post accident sampling system |
| September 27, 1985 | Generic Letter 85-18 - "Operator Licensing Examinations |
| September 27, 1985 | Letter from licensee transmitting License Event Report concerning auxiliary feedwater system |
| September 27, 1985 | Letter from licensee transmitting revised emergency plan implementing procedures |
| September 30, 1985 | Letter from licensee regarding proposed changes to FSAR incorporating results of main steam line break analysis |
| September 30, 1985 | Letter from licensee concerning inadvertent deboration event analysis |
| September 30, 1985 | Letter from licensee requesting extension from implementation deadline of November 30, 1985 deadline and justification for interim operation regarding environmental qualification of hydrogen recombiners |
| September 30, 1985 | Letter from licensee confirming that small break LOCA spectrum conforms to criteria of 10 CFR 50.46 and bounded by limiting large LOCA |



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| September 30, 1985 | Letter from licensee providing description of program to review proposed changes to plant and potential impact on Chapter 15 analysis |
| September 30, 1985 | Letter from licensee transmitting proposed revision to FSAR on quality assurance |
| October 2, 1985 | Letter to licensee regarding auxiliary spray system (50.54(f)) |
| October 3, 1985 | Letter from licensee forwarding specific limiting large break LOCA reanalysis results |
| October 7, 1985 | Letter from licensee requesting exemption from 10 CFR part 50, Appendix J for Units 2 and 3 |
| October 7, 1985 | Letter from licensee forwarding equipment qualification summary sheet |
| October 7, 1985 | Transmittal to licensee of Bi-weekly Sholly Notice dated September 11, 1985 |
| October 9, 1985 | Site visit to confirm Unit 2 security program |
| October 11, 1985 | Letter from licensee transmitting Revision 1 to "Offsite Dose Calculation Manual" |
| October 12, 1985 | Letter from licensee advising of schedule for final implementation of ATWS rule requirements |
| October 15, 1985 | Letter from licensee concerning auxiliary pressurizer spray system |
| October 16, 1985 | Letter from licensee transmitting request for license amendment on environmental qualification |
| October 17, 1985 | Letter from licensee forwarding application regarding sale and leaseback financing transaction by Public Service Company of New Mexico |
| October 17, 1985 | Letter from licensee requesting authorization to use ASME code N-403 |
| October 17, 1985 | Letter from licensee forwarding FSAR Chapter 14 change for Units 2 and 3 |
| October 17, 1985 | Letter from licensee transmitting preoperational radiological monitoring program report for 1979-1984 |



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| October 18, 1985 | Letter to licensee transmitting final draft of technical specifications for Unit 2 |
| October 22, 1985 | Letter from licensee regarding proposed enhancements to the Chemical and Volume Control System |
| October 22, 1985 | Notice of ACRS Subcommittee meeting - November 7, 1985 (Unit 1) |
| October 23, 1985 | Letter from licensee regarding Unit 2 equipment required to be seismically qualified not identical to previously identified Unit 1 equipment |
| October 24, 1985 | Letter to licensee requesting more detailed explanation of how analysis performed on effects of piping weld corrosion on spray pond performance and why results of analysis demonstrate that structural integrity of system will be maintained |
| October 24, 1985 | Letter from licensee transmitting information on Post Accident Sampling System |
| October 31, 1985 | Letter to licensee advising that procedure for Boron Mixing and Natural Circulation Cooldown Test is acceptable for Unit 1 |
| November 1, 1985 | Issuance of Notice of Consideration of an Application for approval of a Proposed Transfer of Interests Under a Proposed Sale and Leaseback Transaction, Proposed no Significant Hazards Determination, Solicitation of Comments and Opportunity for Hearing (Published 11/5/85, 50 FR 45955) |
| November 1, 1985 | Letter from licensee forwarding changes to final draft Technical Specifications for Unit 2 |
| November 4, 1985 | Letter from licensee forwarding results of steam generator tube rupture with loss of offsite power and schedule for implementation of modifications to Auxiliary Pressurizer Spray System |
| November 6, 1985 | Letter from licensee concerning spray pond piping system |
| November 8, 1985 | Letter from licensee forwarding schedule for implementing remaining NUREG-0737, Supplement 1 requirements |



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| November 13, 1985 | Letter from licensee requesting partial schedular exemption to General Design Criterion 4 for Unit 1 |
| November 14, 1985 | Letter from licensee forwarding listing of correspondence affecting FSAR, for which an update will not be issued prior to issuance of Unit 2 license |
| November 15, 1985 | Letter from licensee providing updated request for partial schedular exemption to General Design Criterion 4 for Unit 1 |
| November 18, 1985 | Letter to licensee from Commission Secretary advising that extension has been granted for time to complete environmental qualification of electrical equipment from November 30, 1985 to March 30, 1986 (Units 1 and 2) |
| November 19, 1985 | Letter from licensee requesting change to Technical Specifications to allow performance of required environmental qualification modifications to hydrogen recombiners |
| November 20, 1985 | Issuance of Environmental Assessment and finding of No Significant Impact (re proposed exemption to General Design Criterion 4 for Unit 1) |
| November 20, 1985 | Letter from licensee concerning schedule for remaining Unit 1 power ascension testing with respect to Unit 2 licensing |
| November 22, 1985 | Issuance of Partial Schedular Exemption from a portion of General Design Criterion 4 for Unit 1 |
| November 22, 1985 | Letter from licensee advising of readiness to load fuel |
| November 22, 1985 | Letter from licensee transmitting Justification for Interim Operation of Reactor Coolant Gas Vent System |
| November 25, 1985 | Letter to licensee requesting additional information on auxiliary spray and pressurizer vent systems |
| November 25, 1985 | Letter from licensee responding to request concerning auxiliray spray and pressurizer vent systems |
| November 27, 1985 | Letter from licensee transmitting Justification for Interim Operation for Containment Water Level Instrumentation |



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| November 27, 1985 | Letter from licensee transmitting Technical Specifications certification with changes |
| November 27, 1985 | Letter from licensee with updated DCRDR and HED information |
| November 27, 1985 | Letter from licensee addressing natural circulation test schedule and Unit 2 exceeding 5% |
| November 29, 1985 | Letter from licensee responding to staff questions on potential cracking of charging pump casing block |
| November 29, 1985 | Letter from licensee transmitting Justification for Interim Operation of emergency lighting system |
| November 29, 1985 | Letter from licensee concerning operability schedule of radiation monitors |
| December 2, 1985 | Letter from licensee transmitting Technical Specifications certification, accepting staff comments |
| December 4, 1985 | Letter from licensee addressing charging pump operability and venting commitments |
| December 5, 1985 | Letter from licensee with revised commitments on charging pump operability and venting |



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## APPENDIX E

### Abbreviations

| | |
|---|---|
| ANPP | Arizona Nuclear Power Project |
| APS | Arizona Public Service |
| CHR | containment heat removal |
| DCRDR | detailed control room design review |
| ECC(s) | emergency core cooling (system) |
| GDC | General Design Criterion |
| HED | human engineering discrepancy |
| ID | inside diameter |
| ISI | inservice inspection |
| LOCA | loss-of-coolant accident |
| NRC | Nuclear Regulatory Commission |
| NSSS | nuclear steam supply system |
| PSI | preservice inspection |
| PVNGS | Palo Verde Nuclear Generating Station |
| RG | regulatory guide |
| RHR | residual heat removal |
| SER | Safety evaluation report |
| SSER | Supplement to the safety analysis report |



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# APPENDIX F

## PRINCIPAL CONTRIBUTORS

| Name | Issue |
|---|---|
| M. Ley | Project Management |
| L. Crocker | Shift Staffing |
| B. Elliot | Materials Engineering |
| H. Garg | Equipment Qualification |
| C. Gaskin | Physical Security |
| M. Hartzman | Mechanical Engineering |
| M. Hum | Preservice Inspection |
| M. Lamastra | Radiological Assessment |
| C. Liang | Reactor Systems |
| L. Marsh | Reactor Systems |
| E. Throm | Reactor Systems |
| J. Wing | Post Accident Sampling System |
| R. Wright | Equipment Qualification |
| S. L. Wu | Guide Tube Wear Surveillance |



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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

OFFICE OF THE
SECRETARY

November 18, 1985

Mr. E.E. Van Brunt, Jr.
Executive Vice President
Arizona Nuclear Power Project
Post Office Box 52034
Phoenix, Arizona 85072-2034

Dear Mr. Van Brunt:

By letter dated September 30, 1985, in accordance with 10 CFR 50.49, the Arizona Nuclear Power Project requested an extension of time beyond November 30, 1985, to complete environmental qualification of electrical equipment at its Palo Verde Nuclear Generating Station, Units 1 and 2.

Upon consideration of your September 30 letter the Commission has approved your request.

A license amendment extending the deadline to March 30, 1986 will be issued in the near future.

Sincerely,

Samuel J. Chilk
Secretary of the Commission



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APPENDIX L

UNIT 2 PRESERVICE INSPECTION EVALUATION

PRESERVICE INSPECTION REQUIREMENTS OF 10 CFR 50.55a(g)(3)

1. INTRODUCTION

The staff evaluated similar issues during the review of Palo Verde Unit 1 in Appendix B of SSER 3.

For nuclear power facilities whose construction permits were issued on or after January 1, 1974, 10 CFR 50.55a(g)(3) specifies that the components shall meet the preservice examination requirements set forth in editions and addenda of Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code applied to the construction of the particular component. The provisions of 10 CFR 50.55a(g)(3) also state that the components (including supports) may meet the requirements set forth in subsequent editions of this Code that are incorporated by reference in 10 CFR 50.55a(b) subject to the limitations and modifications listed therein.

On November 13, 1981, the applicant submitted the Palo Verde Unit 1 preservice inspection (PSI) program based upon the 1974 Edition through the Summer 1975 Addenda of the ASME Code. In a letter dated February 5, 1985 the applicant provided a comparison of the PSI programs for PVNGS 1-3. The field PSI programs for Units 2 and 3 have been revised based on the 1977 Edition through the Summer 1979 Addenda except for ASME Code Class 1, 2, and 3 component supports, subject to the limitations contained in 10 CFR 50.55a(b). Specific written relief from Code requirements for Unit 2 was requested and supported by information in letters dated February 5, 1985 and July 5, 1985 pursuant to 10 CFR 50.55a(a)(3). Therefore, the staff evaluations consist of reviewing the applicant's submittals according to the requirements of the 1977 Edition of Section XI through Summer 1979 Addenda and determining if relief from the Code requirements was justified.

As a result of its review of this information, the staff has determined that certain preservice examinations are impractical and the applicant has demonstrated that either (1) the proposed alternatives would provide an acceptable level of quality and safety or (2) compliance with these requirements would result in hardships or unusual difficulties without a compensating increase in the level of quality and safety. The basis for this conclusion is discussed in the subsequent paragraphs of this report.

2. TECHNICAL REVIEW CONSIDERATIONS

2.1 Palo Verde Unit 2 received a construction permit on May 25, 1976. In accordance with 10 CFR 50.55a, the shop preservice inspection and the examination of component supports is based on the 1974 Edition of the ASME Code, Section XI including Addenda through Summer 1975 and the field inspection is based on the 1977 Edition including Addenda through Summer 1979. The ASME first published



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rules for inservice inspection in the 1970 Edition of Section XI. No preservice or inservice inspection requirements existed prior to that date. Because the plant system design and ordering of long lead time components were well underway by the time the Section XI rules became effective, full compliance with the exact Section XI access and inspectability requirements of the Code is not always practical.

2.2 Verification of as-built structural integrity of the primary pressure boundary is not dependent of the Section XI preservice examination. The applicable construction codes to which the primary pressure boundary was fabricated contain material, design, fabrication examination, and testing requirements that, by themselves provide the necessary assurance that the components are capable of performing safely under all operating conditions reviewed in the FSAR and described in the plant design specification. As a part of these examinations, all of the primary pressure boundary full penetration welds were volumetrically inspected (radiographed) and the system was subjected to hydrostatic pressure tests.

2.3 The intent of the preservice examination is to establish a reference or baseline prior to the initial operation of the facility. The results of subsequent inservice examinations can then be compared to the original condition to determine if changes have occurred. If review of the inservice inspection results shows no change from the original condition, no action is required. In the case where baseline data are not available, all indications must be treated as new indications and evaluated accordingly. Section XI of the ASME Code contains acceptance standards that may be used as the basis for evaluating the acceptability of such indications. Therefore, conservative disposition of defects found during inservice inspections can be accomplished even though preservice information is not available.

2.4 Other benefits of the preservice examination include providing redundant or alternative volumetric inspection of the primary pressure boundary using a test method different from that employed during the component fabrication. Successful performance of a preservice examination also demonstrates that the welds can be effectively inspected during the subsequent inservice examination using a similar test method.

In the case of Palo Verde Unit 2, a large portion of the ASME Code-required preservice examinations was performed. The staff has concluded that failure to perform a 100% preservice examination of the welds identified below will not significantly affected the assurance of the initial structural integrity.

2.5 In some cases where the required preservice examinations were not performed to the full extent specified by the applicable ASME Code, the staff will require that these or supplemental examinations be conducted as part of the inservice inspection program. The staff has concluded that requiring these supplemental examinations to be performed at this time (before plant startup) would result in hardships or unusual difficulties without a compensating increase in the level of quality and safety. The performance of supplemental examinations, such as surface examinations, in areas where volumetric inspection is difficult will be more meaningful after a period of operation. Acceptable preoperational integrity has already been established by similar Section III fabrication examinations.



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# 1 INTRODUCTION AND GENERAL DISCUSSION

## 1.1 Introduction

On November 13, 1981, the Nuclear Regulatory Commission (NRC) staff issued its Safety Evaluation Report (SER) relating to the application for licenses to operate the Palo Verde Nuclear Generating Station, Unit Nos. 1, 2, and 3 (PVNGS 1-3); Supplement Nos. 1 through 8 to the SER were issued on February 4, 1982; May 17, 1982; September 23, 1982; March 15, 1983; November 28, 1983; October 31, 1984; December 31, 1984; and June 1, 1985, respectively. The application was submitted by the Arizona Public Service Company (APS or the applicant*) on behalf of itself, the Salt River Project Agricultural Improvement and Power District, Southern California Edison Company, El Paso Electric Company, Public Service Company of New Mexico, Los Angeles Department of Water and Power, and Southern California Public Power Authority. An operating license (NPF-34) restricted to 5% of full power was issued for Unit 1 on December 31, 1984, and a full power operating license was issued on June 1, 1985.

In the SER and its supplements, the staff identified certain issues on Units 2 and 3 for which either further information was required of the applicant or additional staff effort was needed to complete the review of the application. The purpose of this supplement is to update the SER by providing an evaluation of (1) additional information submitted by the applicant since Supplement No. 8 to the SER was issued, and (2) matters that the staff had under review when Supplement No. 8 was issued, specifically those issues that required resolution prior to Unit 2 fuel loading and testing up to 5% of full power.

Each of the following sections of this supplement is numbered the same as the section of the SER (and its supplements) that is being updated and, unless otherwise noted, the discussions are supplementary to and not in lieu of the previous discussions. Appendix A to this supplement is a continuation of the chronology. Appendix E is a list of abbreviations used in this supplement. Appendix F is a list of principal contributors to this supplement, and Appendix L is the Unit 2 Preservice Inspection Evaluation.

## 1.9 Summary of Outstanding Issues

Section 1.9 of Supplement No. 8 contained a list of four issues related to PVNGS Units 2 and 3 that were still outstanding. These issues and six others are listed below, along with the section of this supplement wherein the resolution of each issue is discussed. All issues requiring resolution prior to Unit 2 fuel loading and testing up to 5% of full power have now been resolved.

(1) Guide tube wear surveillance (4.2.5)

(2) Preservice inspection program (5.2.4, 6.6)

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*As of the issuance date of this SSER, APS is the licensee of PVNGS Unit 1 and the applicant of PVNGS Units 2 and 3. However, throughout this SSER, APS is referred to as the applicant.

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(3) Pressurized thermal shock (5.3.3)

(4) Operating experience of shift staffing (13.1.2.3)

(5) ECCS - large break LOCA evaluation model (6.3)

(6) Equipment qualification differences between Unit 1 and Unit 2 (3.10 and 3.11)

(7) Post accident sampling system (22.2)

(8) Pressurizer auxiliary spray capability

(9) Site security plan (13.6)

(10) Radiation monitoring (11.3)

The Detailed Control Room Design Review (DCRDR) issue will remain an open item for both Units 1 and 2 until the staff completes its review of the Supplemental Summary Report dated August 30, 1985. Resolution of this issue does not preclude licensing of Unit 2 and will be addressed in a future supplement to the SER.

Also, there are three issues relating to PVNGS Unit 3 which are still outstanding. These issues are listed below and will be addressed in a future supplement to the SER.

(1) Preservice inspection program

(2) Pressurized thermal shock

(3) Operator experience of shift staffing

1.11 License Conditions

Several conditions from the Unit 1 full power license, issued on June 1, 1985, have not been included in the Unit 2 low power license. These conditions along with their justification for exclusion are listed below.

(1) Emergency preparedness - By letter dated September 25, 1985, FEMA advised the staff of a formal 44 CFR 350 finding of adequate offsite emergency preparedness.

(2) Results of piping vibration test program - The piping vibration test will have been performed on Unit 1.

(3) Radiochemistry laboratory - This condition was required for Unit 1 until Unit 2 is issued an operating license.

The following conditions are included in the Unit 2 low power license; however, they differ from those included in the Unit 1 full power license.

(1) Operating staff experience requirements - See Section 13.1.2.3 of this SSER.



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In cases where parts of the required examination areas cannot be effectively examined because of a combination of component design or current inspection technique limitations, the staff will continue to evaluate the development of new or improved volumetric examination techniques. As improvements in these areas are achieved, the staff will require that these new techniques be made a part of the inservice examination requirements of those components or that received a limited preservice examination.

## 3. EVALUATION OF RELIEF REQUESTS

The applicant requested relief from specific preservice inspection requirements and provided supporting information in letters dated February 5, 1985, and July 5, 1985. The applicant also asked to use the requirements of subsequent editions and addenda of the Code that have been evaluated and found to be acceptable.

Based on the information submitted by the applicant and the staff's review of the design, geometry, and materials of construction of the components, certain preservice requirements of the ASME Code, Section XI have been determined to be impractical. Therefore, pursuant to 10 CFR 50.55a(a)(3), the staff concludes that these preservice requirements are impractical. The justification for this conclusion follows.

### 3.1 Branch Pipe Connection Welds Exceeding Two-Inch Diameter, Examination Category B-J (Relief Request 2)

Code Requirement

A volumetric examination is required of branch pipe connection welds exceeding two-inch diameter. The examination area should include the weld metal, the base metal on the main pipe run and along the branch run for ½ inch from the toe of the weld.

Code Relief Request

The applicant requested relief from performing 100% of the Code-required volumetric examination.

Reason for Request

Because of the geometrical configuration of the weld, piping and nozzle, the applicant requested relief from having to examine 100% of the required material volume. A listing of branch connection welds in this request is included as Table 1. Dimensioned sketches illustrating the volumes that were not examined are included in the applicant's letter dated February 5, 1985. The restricted access from the nozzle forging side, caused by the weld and nozzle configuration, results in less than 100% coverage of the examination volume. Combustion Engineering evaluated possible supplementary or alternative examination and could not identify any practical methods. Table 2 describes the alternative examinations that were considered. Ultrasonic examination were performed to the maximum extent possible on each of the welds from the main pipe run (main coolant pipe). A surface examination was also performed on each of the welds.



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Staff Evaluation:

The staff has evaluated the accessibility and inspectability of the subject branch connection welds and has determined that the examination of these welds, to the extent required by the Code, is impractical because of the design of the piping system and/or limitations in available examination techniques. The staff, agrees with the applicant's basic conclusion that no other practical technique could be applied to meet the Code requirement.

The staff has determined that the applicant has examined the welds to the maximum extent practical. The staff concludes that the Section XI surface examinations and the limited ultrasonic examinations, the examinations performed during fabrication and the hydro-static test demonstrate an acceptable level of preservice structural integrity and that requiring compliance with the specified Code requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

B. Ultrasonic Testing Calibration Blocks (Relief Request 3)

Code Requirement

Paragraph I-3121 of Appendix I to ASME Code Section XI requires that the calibration blocks be fabricated from (1) the component nozzle dropout, (2) the component prolongation, or (3) when this is not possible, blocks may be fabricated from a material included in the applicable examination volumes of the component. The acoustic velocity and attenuation of the block shall be demonstrated to be representative of the component (ASME Code Section XI, Summer 1975 Addenda).

Code Relief Request

Relief is requested to use calibration standards that meet the requirements of later approved editions of ASME Code Section XI.

Reason for Request

The preservice examination of the reactor vessel, pressurizer and steam generators (ASME Code Class 1 and 2 examination categories) was conducted based on the 1974 Edition of Section XI of the ASME Code including addenda through Summer 1975.

ASME Code Section XI edition and the addenda referenced by 10 CFR 50, which will be used for the inservice inspection, require that calibration blocks be fabricated from material of the same specification, product form, and heat treatment as one of the materials being joined. The calibration blocks for the shop examinations are in accordance with this requirement.

Staff Evaluation

Paragraph 10 CFR 50.55a(g)(3)(iv) permits updating to meet the requirements of later approved editions and addenda of ASME Code Section XI subject to the limitations described in 10 CFR 50.55a(b).



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The staff has determined that the ultrasonic testing calibration used during the shop examinations meet the requirements of the regulation and, therefore, the staff concludes that relief from ASME Code requirements is not necessary.

3.3 Circumferential and Longitudinal Pipe Welds with Access Limitations (Relief Request 1)

Code Requirement

A volumetric examination is required of essentially 100% of the longitudinal and circumferential welds and the base metal for ASME Code Class 1 piping (Examination Categories B-J and B-F) in accordance with Code Table IWB-2500-1 and Figures IWB-2500-8,-9,-10 and -11. The Code also requires a surface examination.

A volumetric examination is required of a designated sample of longitudinal and circumferential welds and the base metal for ASME Code Class 2 piping (Examination Category C-F) in accordance with Code Table IWC-2500-1 and Figure Number IWC-2500-7. The Code also requires a surface examination.

Code Relief Request

The applicant requested relief from performing 100% of the Code-required volumetric examination.

Reason for Request

In cases where component configuration or limited access resulted in the examination of less than 100% of the Code required volume, alternative examination techniques were considered, and if feasible, used. (See Supplement 3, Appendix B, Table 4 for a discussion of the alternative examinations that were considered). In letters dated February 5, 1985 and July 5, 1985, the applicant identified the piping system welds with incomplete scans and "one-side" examinations. The applicant's summary, consisting of 18 pages, identifies the welds with examination limitations, the data package number, drawing reference number, calibration standard, scan angles, specific configuration/limitation and whether the obstruction is permanent. Ultrasonic examinations were performed on each weld to the maximum extent possible. A surface examination was also performed on each of the welds.

Staff Evaluation

The staff evaluated the accessibility and inspectability of the welds with incomplete scans and "one-side" examination defined in the applicant's letters dated February 5, 1985 and July 5, 1985. The staff has determined that examination of these welds to the extent required by the Code is impractical because of the design of the piping system and/or limitation of available examinations methods. Typical examples of the types of limitation are:

- Weld reinforcement prevents scanning of the weld subject to examination.
- Instrument nozzles partially obstructing the weld subject to examination.


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- Piping to valve or component welds where examination is possible only from the pipe side of welds.

The staff's review has determined that the applicant has examined the subject welds to the maximum extent practical. The staff concludes that the limited Section XI examination, the volumetric examinations performed during fabrication, and the hydrostatic test demonstrate an acceptable level of the preservice structural integrity and that requiring compliance with the specified Code requirements would result in hardship or unusual difficulty without a compensative increase in the level of quality and safety.

## 4. CONCLUSIONS

Based on its review the staff has determined, pursuant to 10 CFR 50.55a(a)(3), that certain Section XI required preservice examinations are impractical and compliance with the requirements would result in hardships or unusual difficulties without a compensating increase in the level of quality and safety.

The staff's technical evaluation has not identified any practical method by which the existing Palo Verde Unit 2 can meet all the specific preservice inspection requirements of Section XI of the ASME Code. Requiring compliance with all the exact Section XI required inspections would delay the startup of the plant so the applicant could redesign a significant number of plant systems, obtain sufficient replacement components, install the new components, and repeat the preservice examination of these components. Examples of components that would have to be redesigned to meet the specific preservice examination provisions include a significant number of the piping and component support systems. Even after the redesign effort, complete compliance with the preservice examination requirements probably could not be achieved. However, the as-built structural integrity of the existing primary pressure boundary has already been established by the construction code fabrication examinations.

Thus, on the basis of its review and evaluation, the staff concludes that the public interest is not served by imposing certain provisions of Section XI of the ASME Code that have been determined to be impractical. Pursuant to 10 CFR 50.55a(a)(3), the applicant has been allowed relief from these requirements that are impractical to implement and would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety.


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Table 1 Branch nozzle connections to main coolant piping

| Area Design | Description |
|---|---|
| Exam Zone 6 | |
| Item 8 | Surge nozzle-to-pipe weld |
| Item 9 | Shutdown cooling nozzle-to-pipe weld |
| Exam Zone 7 | |
| Item 8 | Shutdown nozzle-to-pipe weld |
| Exam Zone 9 | |
| Item 8 | Safety injection nozzle-to-pipe weld |
| Item 9 | Spray nozzle-to-pipe weld |
| Exam Zone 13 | |
| Item 8 | Safety injection nozzle-to-pipe weld |
| Item 9 | Charging nozzle-to-pipe weld |
| Exam Zone 15 | |
| Item 8 | Safety injection nozzle-to-pipe weld |


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Table 2 Alternative Examinations that were Evaluated FOR PRIMARY PIPING BRANCH CONNECTIONS

| Exam Method | Limitation |
|---|---|
| RT | Not practical for ASME Code Section XI exams because of loss of sensitivity as a result cladding, double wall thickness, orientation of inservice defects, clouding of film due to existing radiation, etc. |
| UT | |
| 1. Angles other than | |
| 30° | The use of 1/2V 30° exam would not improve the amount of the weld or weld required volume that is presently being examined. |
| 70° | While the 1/2V 70° exam would provide a small percentage of improvement, it would still not address the majority of the area not presently examined. Additionally, the calibration blocks that are presently in use of primary pipe are not constructed to allow for a 70° calibration. |
| 2. The use of full V and 1-1/2V exams | Full V and 1-1/2V exams are not recommended because of problems caused by the stainless steel cladding. Examples: beam dispersion, beam and mode conversion for all angles that would yield an uncalibrated beam passing the 1/2V of the beam travel. Because of the nozzles, parallel surfaces are not present so beam location is undeterminable. |
| 3. Scanning on the radius of the weld | Because of the nature of the nozzle-to-pipe blended welds and the nozzle-to-pipe relating a continuous radius, if not maintained, this condition will cause uncoupling or lift off of the transducer or wedge, and will result in angles other than those calibrated for this. This would pertain to surface-conforming fixtures as well. |
| 4. Scanning on the barrel of the nozzle | While this exam would place a sound beam through more of the weld required volume, it would not increase the amount of the weld that is examined. Further, angle exams on the barrel side would not result in the desired beam angle with respect to the weld as per appendix of the Code. The calibration standards that are currently in use are not suitable for performance of this type of examination. |



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Supplement No. 9 to the Safety Evaluation Report for the application filed by Arizona Public Service Company, et al, for licenses to operate the Palo Verde Nuclear Generating Station, Units 1, 2 and 3 (Docket Nos. STN 50-528/529/530), located in Maricopa County, Arizona has been prepared by the Office of Nuclear Reactor Regulation of the Nuclear Regulatory Commission. The purpose of this supplement is to update the Safety Evaluation Report by providing an evaluation of (1) additional information submitted by the applicants since Supplement No. 8 was issued and (2) matters that the staff had under review when Supplement No. 8 was issued, specifically those issues which required resolution prior to Unit 2 fuel loading and testing up to 5% of full power.

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